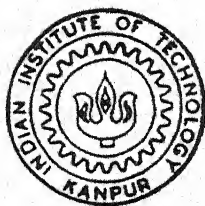


# A PROBABILISTIC ANALYSIS OF PHWR FUEL ELEMENT BEHAVIOR

by  
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**DEPARTMENT OF NUCLEAR ENGINEERING AND TECHNOLOGY  
INDIAN INSTITUTE OF TECHNOLOGY KANPUR**

**JULY, 1994**

# A PROBABILISTIC ANALYSIS OF PHWR FUEL ELEMENT BEHAVIOR

A Thesis Submitted  
in Partial Fulfillment of the Requirements  
for the Degree of  
MASTER OF TECHNOLOGY

by

A. SANDEEP

to the

NUCLEAR ENGINEERING & TECHNOLOGY PROGRAMME  
INDIAN INSTITUTE OF TECHNOLOGY , KANPUR

JULY, 1994

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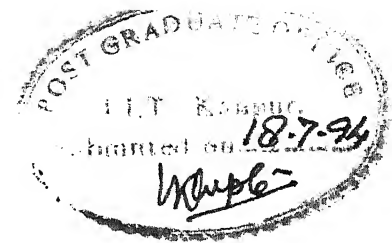
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## CETIFICATE

This is to certify that this work on 'A Probabilistic Analysis Of PHWR Fuel Element Behavior' by A.Sandeep has been carried out under our supervision and has not been submitted elsewhere for the award of a degree.

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- A. Sandeep

## ABSTRACT

Probabilistic analysis of fuel element behavior in a reactor, it has been indicated, can play a very important role in reactor safety analysis. Such analysis is much more realistic than the conservative and oversafe (deterministic) worst case analysis. The present work describes and evaluates two statistical methods - Monte Carlo simulation and orthogonal array experiments - in a probabilistic study of PHWR fuel element behavior.

The important parameters having a bearing on fuel element behavior are gap conductance, maximum clad temperature, fuel center-line temperature and fission gas internal pressure. These output parameters are dependent upon various input parameters including as-fabricated parameters (dimensions and mass and density of fuel), modeling parameters (fission gas release, swelling, cladding creep and cracking of fuel) and power history data. These input parameters have a distribution around their nominal values which reflects in the distribution of output parameters. The worst case values (using worst combination of values at tolerance limits) give very conservative values of output parameters, whereas their statistical distributions give a more realistic and versatile look at the output parameters and hence fuel rod behaviour.

The procedure involves two steps. The first step includes developing statistical distributions of input parameters based on the tolerance limits, preparing models for various modeling and output parameters of interest and finally performing Monte Carlo simulations to obtain output distributions for the desired variables. The second step comprises the conduct of statistically designed experiments with the input parameters set at certain appropriately selected levels and estimating the factor effects on

the output.

Realistic conclusions about fuel element behaviour may be drawn from the output parameter distributions thus developed and the factor effects estimated. For example, the probability distributions determined can help quantify the fraction of fuel center-line temperature exceeding the melting point, the fuel element internal pressure exceeding the external coolant pressure, the probability of pellet-clad interference, etc. Such statistical information can be then compared with the worst case values to quantify the degree of conservatism in the latter. The factor effects estimated provide valuable insight to the fuel system designer as to the direction of and the extent to which the input (design) parameters should be altered to reduce these critical probabilities.

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## CHAPTER.1

### INTRODUCTION

Probabilistic design methods are being increasingly used in different fields of nuclear engineering with a view to ensure that nuclear power plants are designed and operated to high standards of safety. Probabilistic risk assessment, which considers the safety aspects of the entire nuclear power plant, is a well known example. The approaches based on statistical combinations of uncertainties are also becoming more and more important in the design and analysis of components and subsystems -- particularly those which have a bearing on reactor safety .

Perhaps the most important operational concern both under steady state and overpower transients as well as accidental conditions is the performance of fuel rods in the reactor core. The fuel rod behaviour is influenced by many parameters related to the design, fabrication, material properties and operation parameters such as burn-up, power history, etc. In addition , many modelling parameters related to pellet restructuring and cracking, densification, swelling, gaseous fission product migration and build-up, pellet-clad interaction and cladding creep, etc. affect the performance of the fuel pins. Since the failure modes under accident conditions are strongly dependent on the state of the fuel elements at the time of the initiation of the accident, considerable emphasis must be placed on thorough investigation of the steady state behaviour of the in-core fuel .

The various parameters mentioned above show a large scatter within the specified tolerances as a result of random variation or systematic bias during the fabrication process, or, due to uncertainties in the modelling process . At present , the common practice is to use a combination of superimposed unfavorable

limits (the worst case dataset ) in the fuel rod design calculations and performance prediction . The purpose of the present study is to initiate a systematic statistical analysis of the modelling process as well as the fuel rod fabrication quality control data and evaluate the resulting performance uncertainties.

The study uses the available quality control data for PHWR fuel rod fabrication processes [11]. This includes the data related to pellet density , pellet clad dimensions ( pellet diameter , inner clad diameter , outer clad diameter , clad length ) , and powder characteristics ( average grain radius of the fuel grains ). Input distribution functions for the above were first generated . Then , the uncertainties in the modelling parameters were considered . Important examples are the parameters related to fission gas release , fuel pellet swelling , and thermal expansion of the cladding . Due to considerable information gaps and also the fact that many of these are areas of current research , some parametric values have been considered for some of these to cover the range of interest. In addition , cumulative burn-up , linear thermal power rating and certain material properties ( thermal conductivities , thermal expansion coefficients , etc.) have been considered as fixed. Further one-dimensional models have been used to evaluate the output parameters that characterize the performance of the fuel rod, namely, fuel center-line and inner clad temperatures, gap heat transfer coefficient or gap conductance , and fission gas pressure.

The distribution functions of the input parameters are used in performing the Monte Carlo simulations to determine the dependency of the location and spread characteristics of the key performance variables of the fuel rods. Various deductions about performance are then made on the basis of the performance characteristics like how much percentage of fuel rods will melt

near their center, the percentage of cases in which the internal fission gas pressure is greater than the coolant pressure, the fraction of the pellets in interference with clad, and the most probable and nominal values of the performance variables.

An important aspect, especially for the safe design of the fuel system, is knowing quantitatively how the different design factors - pellet diameter, cladding dimensions, powder characteristics, etc. - influence fuel element behavior individually and collectively ( as their influence may interact ). To this end orthogonal array-based statistical experiments are conducted and the design factor effects estimated.

## CHAPTER 2

### MODELING AND OUTPUT PARAMETERS

#### 2.1 INTRODUCTION

During reactor operation , fission products are continuously formed . Accumulation of the fission products can drastically change the properties and dimensions of the solid fuel . About ten % of the fission products are rare gases , and the remainder are metallic and non-metallic solids . The rare gases will not combine with , or dissolve in , the matrix and therefore they largely contribute to swelling and pressure buildup. The other fission products will combine or alloy with the remaining fuel atoms .

Radiation induced processes alter almost every measurable property of a solid fuel . Most of these changes are not of primary concern from the standpoint of reactor operation with the exception of dimensional changes that accompany fissioning. These alterations in dimensions/shape can lead to cladding failure , interference with coolant flow, etc. These effects become more severe as the operating temperature is increased .

Further , reactor operates under severe conditions like high temperature , large coolant pressures, and high irradiation . These conditions make it more susceptible to creep , thermal stresses, and consequent cracking of the fuel element. Hence the performance of the reactor is prone to serious damage if these conditions become worse .

#### 2.2 FISSION GAS RELEASE MODEL

The fission products most important with respect to their effect on the stability of reactor fuel elements are the stable fission products. For this purpose , the tabulation of Katcoff has

been used [1]. The stable fission products consist predominantly of metal isotopes which would be expected to exist substitutionally or interstitially as cations in the oxide lattice . A much smaller fraction of the products , consisting of , Se, Te, I, Br would be expected to exist as anion impurities substitutionally or interstitially . The third class of fission products , the most important to oxide fuel element behaviour , consists of the noble gases xenon and krypton which are formed in amounts corresponding to about 0.25 noble gas atoms per U-235 nucleus fissioned . Their method of incorporation in the oxide lattice is little known ; however, their ability to escape at free surfaces of the oxide indicates that they are not tightly bound in the lattice .

Three methods of their escape from the oxide lattice have been identified . The first involves escape by recoil from those atoms fissioned within the recoil range in the solid . This mechanism is most important in the case of irradiations performed at low temperature or in the case of irradiation of fine particles such as dispersion fuels . The instantaneous release rate by this mechanism is independent of the nature of the isotope or of the irradiation temperature and depends only on its fission yield and recoil range in the solid and on the external surface area . The second method of escape is by solid state diffusion . This is most important for the case of bulk oxides operating at elevated temperatures characteristic of the operating temperature of the fuel elements . For very high temperatures (  $> 2000$  K ), other release mechanisms are effective . Fission gas bubbles precipitated inside the fuel as well as the pores in the material have enough mobility to enable them to migrate over large distances across temperature gradients . Gas release from these cavities take place when they come across cracks and channels in the fuel that are connected to the inner empty volume . However ,

this mechanism becomes important only for accidental conditions or for very high operating temperatures .

Thus for steady state operation , gas release by diffusion assumes the most significant role. On the assumption that fission gases are uniformly generated over the entire volume , that they donot decay sharply in time , and that the medium is homogeneous and isotropic , the diffusion of stable fission gases is described by the Fick's law :

$$\frac{dc}{dt} = \frac{1}{r^2} \frac{d}{dr} \left[ \frac{r^2 D}{dr} \frac{dc}{dr} \right] \quad ( 2.1 )$$

where,

$c(r,t)$  is the gas atom concentration,

$B$  is their rate of generation by fission, and

$D$  is the diffusion coefficient .

By integrating the eq.( 2.1 ) over a sphere of radius  $a$  (an idealised  $UO_2$  grain ), with boundary and initial conditions  $\therefore c=0$  for  $r=a$  and  $c=0$  for  $t=0$  and defining the release fraction ,  $f$ , as the ratio of the quantity that leaves the fuel grain by diffusion to the total quantity generated by fission, one obtains [3]:

For  $\pi Dt/a^2 < 1$ ,

$$f = 4 \left[ \frac{D^2 t}{\pi a^2} \right]^{1/2} - 1.5 \frac{D t}{a^2} \text{ , and}$$

for  $\pi Dt/a^2 > 1$ ,

$$f = 1 - \frac{a^2}{15 Dt} + 6 \frac{a^2}{\pi^2 Dt} \exp \left[ - \frac{\pi^2 D t}{a^2} \right] \quad ( 2.2 )$$

where  $t$  is the irradiation duration.

The diffusion coefficient is usually obtained empirically. From the studies made by a number of investigators [2], the following formula has been obtained for the diffusion coefficient  $D$ :

$$\log D = -13.5 - \left[ \frac{10000}{T} - 5 \right] * 0.6 \quad ( 2.3 )$$

where T is the temperature at which diffusion coefficient is to be evaluated.

Since there is a large temperature gradient across the fuel element and fractional release is a strong function of the temperature ( through diffusion coefficient ) , to estimate the fractional release of fission gases from the fuel elements a three point approximate Simpson's rule is used [3]. According to the rule , the fractional release from the entire fuel element ,  $f_{tot}$  , is given by

$$f_{tot} = \frac{1}{6} ( f_1 + 4 f_2 + f_3 ) \quad ( 2.4 )$$

where ,

$f_1$  is the fractional release at fuel centre-line temperature,  
 $f_2$  is the fractional release at mean fuel temperature, and  
 $f_3$  is the fractional release at fuel surface temperature.

### 2.3 SWELLING OF THE FUEL ELEMENT

Swelling is the dimensional instability of fuel materials caused by various reasons during operation/irradiation of fuel. Fuel swelling originates from four main sources : positive effects due to solid fission product accumulation , thermal expansion , growth and movement of fission gas bubbles , and the negative effect due to densification.

Since the atomic volume of the fission product atoms is greater than that of a uranium atom, a corresponding increase in the fuel element volume results. These solid fission products take up positions close to the end of their recoil range in the material or diffuse to a more stable configuration , depending on

the temperature. This accumulation of the solid fission products leads to an increase in volume (or swelling ).

The operating temperatures of the fuel element are much higher than the assembly or fabrication temperature so thermal expansion of the fuel element results in swelling of the fuel.

If the temperature is high enough to permit diffusion of the inert-gas fission products (which is of course there inside the fuel elements ) gas bubbles form by both heterogeneous (at structural inhomogeneities) and homogeneous nucleation. The stable nuclei then grow by capturing additional gas atoms. Swelling, under irradiation, due to collection of fission gases into pores and their movement, would be the resultant of various complex processes. These include rate of production of fission gas, rate of diffusion of fission gas in the matrix, rate of nucleation and growth of pores and extent of chemical and structural inhomogeneity of the fuel, creep rate of the matrix and rate of metallurgical instabilities or structural changes (such as dislocation motion, recrystallization, grain growth or phase change ).

Finally, irradiation leads to the removal of sintering pores in the fuel elements and hence diminishing of the initial porosity. This is a negative effect and leads to densification or negative swelling.

Reference[4] gives an evaluation of the fuel swelling vs local burn-up. In the central zones of the fuel, where temperature is high ( $> 1950$  K), the swelling rate is lower because the fuel material retains negligible quantities of gas( there is a high release in this zone ). In the intermediate temperature range  $1270-1970$  K , where the fuel plasticity and gas atom mobility are considerable and gas release is lower, the fuel swelling has maximum values. At lower temperature fuel swelling again falls as

fuel densification becomes important.

Net swelling of the fuel element is the result of all the four phenomena. These phenomena themselves are a result of complex processes which are difficult to quantify. In addition, swelling can be accommodated in the porosity of the fuel. Therefore, modeling of swelling has not been fully possible and it is an area of current research. Several studies, including references[5] and [6] have found that the swelling rate of irradiated  $\text{UO}_2$  at operating temperatures is 0.7% change in volume per  $10^{20}$  fissions per  $\text{cm}^3$ . ( 2% per 10,000 MWD/ton of U ) under certain conditions. To account for this lack of accurate information about swelling in the fuel element and due to an important role of fuel swelling in fuel element performance, three different swelling rates of 2%, 3%, and 4% have been taken into consideration in the calculations presented in this work.

Volume,  $V$ , after swelling is given by,

$$V = V_0 (1 + f_s)$$

where,

$V_0$  is the fabricated volume of the fuel material, and  
 $f_s$  is the fraction of swelling .

Assuming that fuel swelling is equally distributed in the three directions, the fractional increase in the diameter can be taken as one third of the swelling. The diameter of the fuel element after swelling,  $d_p$ , then becomes ,

$$d_p = d_{p0} (1 + f_s/3)$$

where  $d_{p0}$  is the fabricated fuel element diameter.

## 2.4 THERMAL EXPANSION OF THE CLADDING/SHEATH

The cladding is very thin, therefore, linear expansion of the

cladding is justified. So the expanded values of the inner clad diameter(  $d_i$  ) and outer clad diameter , $d_o$  , are given by,

$$d_i = d_{i0} (1 + \alpha_c \cdot \delta T) \quad ( 2.5 )$$

$$d_o = d_{o0} (1 + \alpha_c \cdot \delta T) \quad ( 2.6 )$$

where

$d_{i0}$  is the fabricated inner clad diameter,

$d_i$  is the inner clad diameter after thermal expansion,

$d_{o0}$  is the fabricated outer clad diameter,

$d_o$  is the outer clad diameter after thermal expansion,

$\alpha_c$  is coefficient of linear thermal expansion of cladding, and

$\delta T$  is the temperature rise from assembly to operation.

## 2.5 CLADDING CREEP

Creep is a phenomenon in which a body subjected to stress for long time and at high temperature undergoes a continuous, irreversible deformation. Creep presents three stages : primary creep, during which the creep rate slows down; secondary or steady-state creep, during which the creep rate essentially stays constant and tertiary creep in which the creep rate soars shortly before failure occurs.

Methods have been evolved for obtaining creep. It is postulated to consist of two terms; the first term describes the thermal creep, whereas the second term describes the irradiation-induced creep. The strain rate is given as follows [7]:

For zircaloy cladding, creep rate is given as,

$$\frac{d \epsilon}{d t} = 5.129 \cdot 10^{-29} \phi (\sigma + 7.252 \cdot 10^2 \exp(4.967 \cdot 10^{-8} \sigma)) e^{-10000/RT} t^{-1/2} \quad ( 2.7 )$$

where

$\phi$  is the fast neutron flux, per  $\text{cm}^3$

$\sigma$  is the applied stress,  $\text{N/m}^2$

R is the universal ideal gas constant, J/K

t is the duration of stress application, h and

T is the operating temperature, K.

Similar expression is available for creep rate of fuel material.

In a reactor, cladding materials are required to withstand, without rupture, the strains imposed by the release of fission gases from the fuel during irradiation. In an operating reactor, the pressure buildup due to fission gas release is both time and temperature dependent. The resistance of a cladding material under those conditions is studied by examining creep.

However, creep depends upon many other complex phenomena like swelling, pellet clad interaction and structural changes of the fuel, which themselves are not very well understood. In addition, creep is more important at large stresses and at high operating temperatures( e.g., under defect or accidental conditions). Its value under normal operating conditions is quite small. Therefore, in the present study the effect of creep is neglected in evaluating the performance variables of the fuel element.

## 2.6 STRUCTURAL CHANGES AND OTHER EFFECTS [8, 2, 3]

(a) Fuel ratchetting: References suggest that in a fuel element held vertically, operating with extensive central melting, the molten fuel would slump to the bottom of the element where it would solidify during shutdown. Subsequently a fast start-up (accompanied by peak heat ratings near the bottom of the element ) could cause the expanding fuel to strain the sheath/clad radially rather than expand axially to fill the solidification void. Several unexpected failures were ascribed to this type of mechanism , and the adoption of a slow, programmed reactor startup seemed to cure the problem. Another possible solution may be to

compartment the fuel element along its length, though there may be drawbacks to this. There is hardly any problem from ratchetting of molten fuel in a reactor in which the fuel is horizontal.

(b) Radial relocation: The fuel element undergoes severe conditions, like high thermal gradients and thermal stresses, structural changes and irradiation, during operation. Under these conditions, the element may crack especially near the end of its life. These cracked elements must certainly relocate themselves in contact with the inner surface of the cladding. Under severe cracking, the gap volume is distributed throughout the cross-section of the element. This results in enhanced fission gas release and hence increased fission gas pressure; also, the gap heat transfer coefficient and temperature distribution are affected.

(c) Longitudinal ridges: It is a type of distortion in which either wrinkles develop in the sheath or ridges develop along the length. Longitudinal ridging occurs when the clearance between the fuel and clad exceeds a certain critical value (dependent on clad dimensions, properties and the external pressure), and therefore is more likely to occur at zero or low power.

Longitudinal ridging can be controlled by setting a maximum permissible diametrical clearance on fabrication. This would be lower if the clad thickness is reduced or the operating pressure increased. Fuel that expands at power to strain the clad will not contract back to its former diameter, since differential contraction between inner and outer regions of the pellets form shrinkage cracks at the center.

(d) Ovality: It is a type of fuel distortion in which a general ovality develops in the cross section of the fuel element. Development of ovality can occur when there is a void present at the center of the fuel, and when the compressive strength of fuel

plus clad is insufficient to withstand the external coolant pressure. In the extreme case ovality emboldens ridging. The tendency to formation of ovality may depend on fuel diameter, on element power output and on the voidage existing at power. Since shrinkage cracks or a solidification void will form when an element operates at a lower power output than its previous maximum, there is a possibility that such a fuel element may develop ovality. However, the internal gas pressure, since it generally increases throughout the element lifetime, will tend to support the cladding against collapse.

(e) Fuel structure changes:

During reactor operation, the following sequence of restructuring zones are observed across a fuel cross-section: a peripheral zone having a structure practically identical to the original one, an equiaxial-grain zone a columnar grain zone and, for very large linear powers, a central hole.

The equiaxial grain growth is determined by the natural tendency of the total free energy of the surface to decrease, that translates in a diminishing number of grains and, consequently, of the total grain area. In the process, intergranular surfaces shift in the direction of curvature, the large grains with concave surfaces growing on account of this and including the smaller grains with convex surfaces. The pores and impurities on intergranular surfaces diminish their movement rate so that the growth rate is related to material transfer across the pores.

The columnar grain shape is explained by lenticular pore migration through temperature gradients, accompanied by material transport in the opposite direction. During columnar grain growth the pores migrate in the direction of the thermal gradient and accumulate near the fuel center, generating a central hole.

(f) Others

In addition, various other factors come into picture. These include the change of mechanical properties as a result of direct interactions with the fast neutrons and alpha particles, the embrittlement due to hydrogen absorption in Zircaloy, void formation etc.

These structural and other effects are very difficult to quantify fully. However, their influence is greatest under high operating temperature, frequent and large power variation, and assembly asymmetry of fuel location within the cladding, and also depend on the quality of fuel material, and burn-up etc. For our case involving steady-state operation at constant power and normal operating temperature, the effect of these factors can be neglected.

## 2.7 OUTPUT OR PERFORMANCE PARAMETERS

### (a) Gap Conductance or Gap Heat Transfer Coefficient

Considering one-dimensional heat transfer radially along the gap (in cylindrical coordinates), the heat flow rate  $q$ , along the gap is given by,

$$q = -k_g A \frac{dT}{dr}, \quad W \quad (2.10)$$

where

$k_g$  is the thermal conductivity of the gap,  $W \, m^{-1} K^{-1}$

$A$  is the area across which heat flows,  $m^2$

$T$  is the temperature, and

$r$  is the radial direction parameter.

Integrating this equation and applying the limits of integration, the temperature drop across the gap ( $\delta T$ ) is given by,

$$\delta T = \frac{q'}{2\pi k_g} \ln (d_i/d_p) \quad (2.11)$$

where

$q'$  is the linear heat flow rate,  $W\ m^{-1}$   
 $d_p$  is the pellet or fuel element diameter, and  
 $d_i$  is the inner clad diameter.

If  $h_g$  denotes the gap conductance or gap heat transfer coefficient, then the temperature drop across the gap is,

$$\delta T = \frac{q'}{\pi d_p h_g} \quad (2.12)$$

Comparing eqs.(2.11) and (2.12), the gap conductance is

$$h_g = \frac{2\ k_g}{d_p \ln(d_i/d_p)} \quad (2.13)$$

This gives the first output parameter, viz. gap conductance.

#### (b) Inner Clad Surface Temperature

Assuming a constant heat transfer coefficient from the cladding to coolant, the temperature drop between coolant and clad outer surface  $\Delta T_c$ , is given by,

$$\Delta T_c = \frac{q'}{\pi d_c h_c} \quad (2.14)$$

where

$h_c$  is heat transfer coefficient from cladding to coolant, and  
 $d_c$  is the outer cladding diameter.

Similarly, for one-dimensional heat transfer in radial direction, the temperature drop between inner clad surface and outer clad surface  $\Delta T_i$ , is given by,

$$\Delta T_i = \frac{q'}{2\pi k_c} \ln(d_c/d_i) \quad (2.15)$$

Combining eqs.(2.14) and (2.15), the inner clad surface temperature in terms of coolant temperature  $T_{cool}$ , is given by,

$$T_i = T_{cool} + \frac{q'}{\pi} \left[ \frac{1}{d_c h_c} + \frac{\ln(d_c/d_i)}{2 k_c} \right] \quad (2.16)$$

This gives the second output parameter, viz. inner clad temperature

### (c) Fuel Center-Line Temperature

For a rod-shaped fuel element, it has been found that, in a steady-state condition, a parabolic type temperature distribution sets across it. If the volumetric heat generation rate is  $q''$ ,  $Wm^{-3}$ , then the equation describing the temperature distribution in the fuel element reads,

$$\frac{1}{r} \frac{d}{dr} \left[ \frac{r k_p dT}{dr} \right] + q'' = 0 \quad (2.17)$$

where

$k_p$  is the thermal conductivity of the fuel pellet, and

$r$  is the radial coordinate.

Assuming  $k_p$  and  $q''$  are constant across the cross-section of the pellet, the integration of eq.(2.17) with boundary conditions of the type:  $T(R) = T_s$  and  $(dT/dr)_{r=0} = 0$  results in:

$$T - T_s = \frac{q'' R}{4 k_p} \left[ 1 - r^2/R^2 \right] \quad (2.18)$$

Integrating this equation from the fuel center-line to the fuel surface and assuming thermal conductivity as constant (which is the same as the thermal conductivity at mean fuel temperature), the fuel centre-line temperature,  $T_{cl}$ , in terms of fuel surface temperature,  $T_s$ , is given by,

$$T_{cl} = T_s + \frac{q'}{4\pi k_p} \quad (2.19)$$

Combining eq.(2.19) with eqs.(2.12), (2.14) and (2.15), the fuel centre-line temperature,  $T_{cl}$ , in terms of coolant temperature,  $T_{cool}$ , is given by,

$$T_{cl} = T_{cool} + \frac{q'}{\pi} \left[ \frac{1}{4k_p} + \frac{1}{d_p h_g} + \frac{\ln(d_c/d_i)}{2k_c} + \frac{1}{d_c h_c} \right] \quad (2.20)$$

This gives the third output parameter, viz. fuel center-line temperature.

(d) Fission Gas Internal Pressure in the Fuel Pin

Initially the fuel to clad gap in the fuel rod is filled with a rare gas at low pressure. As the fission or reactor operation proceeds, more and more fission gases are released from the fuel element and accumulate in the gap available. In addition, due to swelling of the fuel element and thermal expansion of the cladding, the fuel to clad gap and the volume available to the gas itself reduces. Both these processes result in the rise of internal fuel rod pressure due to fission gases.

Assuming fission gases to be ideal, from ideal gas law the pressure inside the fuel rod ,p, is given by,

$$p = n R T_g / V \quad ( 2.21 )$$

where

n is the number of moles of gases present in the gap,

R is the universal ideal gas constant,

T<sub>g</sub> is the average gap temperature, and

V is the volume available for the gases in the gap.

Volume available for the gases in the gap is the difference of total volume present in the gap and the total volume of the fuel material in the fuel element. The two are to be corrected for swelling of the fuel and thermal expansion of the cladding. The volume available for the gases ,V, is then given by,

$$V = \pi d_i^2 L / 4 - (1 + f_s) m / d \quad ( 2.22 )$$

where

d<sub>i</sub> is the inner clad diameter,

L is the clad length,

$f_s$  is the fractional swelling of fuel,

$m$  is the mass of the fuel in the fuel element, and

$d$  is the density of the fuel in the element.

Number of moles of fission gases present in the gap depends upon their production rate in the fuel rod (or fuel element) and the fractional release of this produced amount. Also, calculations reveal that percentage of initial rare gases in the fuel to clad gap is close to ten percent at the end of life, so neglecting these number of moles of gases  $n$ , in the gap, is given by,

$$n = 0.25 * f * (\text{number of fissions}) / N_A \quad (2.23)$$

where

$f$  is the fractional fission gas release,

0.25 is the number of stable fission gas atoms per fission,

$N_A$  is the Avagadro number,

number of fissions =  $B * m * 3.125 * 10^{10}$ ,

$3.125 * 10^{10}$  is the number of fissions required for each joule produced,

$B$  is the burn-up, and

$m$  is the mass of U present in each element

Using eqs.(2.22) and (2.23), eq.(2.21) gives the fourth output parameter, viz. fuel rod internal pressure due to fission gases. This completes the presentation of the fuel rod modeling and output parameters.

## CHAPTER 3

### NOMINAL AND WORST CASE DETERMINISTIC ANALYSIS

#### 3.1 INTRODUCTION AND DATA

Most of the input parameters are known with their nominal values and their variation( tolerance). Where the variation is not known, constant values have been used. Further, the material properties available from different sources are generally assigned constant values at appropriate conditions close to operating values. These can be used to obtain the nominal and the worst case data values of the output variables of interest. These include the fractional fission gas release, gap heat transfer coefficient, inner clad surface temperature, fuel center-line temperature, and fuel rod internal pressure due to fission gases. Various input parameters and their tolerances as used in the present work are given in Table 3.1 [ 2, 11, 13]

#### 3.2 CALCULATION OF NOMINAL VALUES

##### (a) Fractional Fission Gas Release

From the linear thermal power rating ( $q'$ ) and burn-up ( $B$ ) as given in Table 3.1 and using nominal values of mass ( $m$ ) and density ( $d$ ), the average time of residence of the bundle,  $t$ , is given by,

$$t = B * m / (q' * l) = 320.86 \text{ days.}$$

If the fuel center-line temperature ( $T_c$ ) is taken as approximately 2000 K and fuel surface temperature ( $T_s$ ) as 700 K [11], the fuel median temperature ( $T_m$ ) becomes 1350 K. From eq.(2.3) we get diffusion coefficients at these temperatures as

$$D(\text{at } T=2000 \text{ K}) = 3.162 * 10^{-18} \text{ m}^2 \text{ s}^{-1}$$

$$D(\text{at } T=1350 \text{ K}) = 1.148 * 10^{-19} \text{ m}^2 \text{ s}^{-1}$$

TABLE 3.1 INPUT DATA

Fuel pellet diameter,	$d_p = 14.27^{+0.04} \text{ mm.}$
Outer clad diameter,	$d_c = 15.22^{+0.04} \text{ mm.}$
Inner clad diameter,	$d_i = 14.40^{+0.0894} \text{ mm.}$ (using nominal values and tolerances of $d_c$ and clad thickness )
Cladding length,	$L = 485.8^{+0.05} \text{ mm.}$
Linear thermal power rating,	$q' = 55.0 \text{ kWm}^{-1}.$
Fuel burnup,	$B = 12,500 \text{ MWD per ton U-metal}$
Mass of U in fuel bundle,	$m = 13.412^{+0.217} \text{ kg.}$
Density of fuel material,	$d = 10,600^{+150} \text{ kgm}^{-3}.$
Number of fuel elements in a fuel bundle	$= 19.$
Average $\text{UO}_2$ grain radius,	$a = 50^{+3} \mu\text{m.}$
Linear thermal expansion coefficient of cladding,	$\alpha = 6.07 \times 10^{-6} \text{ K}^{-1}$
Thermal conductivity of the fuel,	$k_p = 3.1 \text{ Wm}^{-1} \text{ K}^{-1}.$
Thermal conductivity of the gap,	$k_g = 0.05 \text{ Wm}^{-1} \text{ K}^{-1}.$
Thermal conductivity of the cladding,	$k_c = 14.0 \text{ Wm}^{-1} \text{ K}^{-1}.$
Heat tranfer coeff. from coolant to clad,	$h_c = 63,460 \text{ Wm}^{-1} \text{ K}^{-1}.$
Average coolant temperature,	$T_f = 550 \text{ K.}$
Average coolant pressure,	$p_f = 95 \text{ bar.}$
Fuel swelling fractions,	$f_s = 2\%, 3\%, 4\%.$

$$D(\text{at } T=700 \text{ K}) = 8.511 \times 10^{-24} \text{ m}^2 \text{ s}^{-1}.$$

Using the values of  $D$ ,  $t$ , and  $a$  in eq.(2.2), fractional fission gas release are obtained as

$$f_1 = 0.37 \text{ (release at fuel centre-line temperature)}$$

$$f_2 = 0.079 \text{ (release at median fuel temperature)}$$

$$f_3 = 0.0007 \text{ (release at fuel surface temperature)}$$

From the three-point model eq.(2.4), the nominal value of fractional fission gas release,  $f$ , is obtained as

$$f = 0.114 \text{ or } 11.4\%.$$

#### (b) Gap Heat Transfer Coefficient

Swelling plays an important role in determining the heat transfer coefficient apart from pellet diameter and inner clad surface diameter. According to our model for swelling, we have three discrete values of swelling. For calculating nominal values 3% swelling is chosen. Fission gas release enters only indirectly through gap thermal conductivity which depends upon the composition of the gases and a suitably chosen value is assigned to it.

Applying swelling and thermal expansion to the nominal value of pellet diameter,  $d_p$ , and inner clad diameter,  $d_i$ , respectively,

$$d_p = 14.4127 \text{ mm, and}$$

$$d_i = 14.4306 \text{ mm.}$$

From eq.(2.11) the nominal value of gap heat transfer coefficient at 3% swelling is,

$$h = 5581.3 \text{ W m}^{-2} \text{ K}^{-1}.$$

### (c) Inner Clad Surface Temperature

Only inner clad diameter and the outer clad diameter are the important input parameters here. Applying thermal expansion to these through eqs. (2.5) and (2.6),

$$d_i = 14.4306 \text{ mm, and}$$

$$d_o = 15.2524 \text{ mm.}$$

Using these values and other constants in eq. (2.14), the nominal value of inner clad surface temperature  $T_i$  is,

$$T_i = 602.72 \text{ K}$$

This is independent of fuel swelling and fractional fission gas release.

### (d) Fuel Center-line Temperature

It depends upon fuel swelling as well as thermal expansion of the cladding (which affect pellet diameter, inner clad diameter and outer clad diameter). However, the fractional fission gas release enters only indirectly through gap heat transfer coefficient. Using the values calculated earlier and assuming a swelling of 3%, the nominal value of fuel center-line temperature  $T_{cl}$  is,

$$T_{cl} = 2232.2 \text{ K}$$

This is well below the melting point of the fuel material which is approximately 3023 K.

### (e) Fuel Rod Internal Pressure

This is strongly dependent on swelling of the fuel, fractional fission gas release from the fuel element, thermal expansion of the cladding and burn-up. For the nominal values of the different variables a swelling of 3%, and a fractional fission gas release of 11.4%, eq. (2.23) gives,

number of fissions(end-of-life)= $2.3824 \times 10^{24}$ .

Number of moles of fission gases,  $n=1.127 \times 10^{-2}$  mol.

Volume available for this gas at nominal values is given by eq.(2.22) as:

$$V=1.64 \times 10^{-6} \text{ m}^3.$$

Then eq.(2.21) gives the nominal value of fuel rod internal pressure due to fission gases ,p, at 3% swelling as,

$$p=36.65 \text{ bar}.$$

### 3.3 WORST CASE ANALYSIS

#### (a) Fractional Fission Gas Release

Taking the worst case of the limits imposed on the variables determining fractional fission gas release, average fuel grain radius  $a=47 \mu\text{m}$ . Using the other variables as in Section 3.2(a), from eq.(2.2) the fractional fission gas releases are obtained as

$$f_1(T=2000 \text{ K})=0.39.$$

$$f_2(T=1350 \text{ K})=0.0835.$$

$$f_3(T=700 \text{ K})=0.00074.$$

From three-point model, eq.(2.4), the net fractional fission gas release from the fuel element for worst case is,

$$f=0.1208 \text{ or } 12.08\%.$$

#### (b) Gap Conductance or Gap Heat Transfer Coefficient

Maintaining the swelling to be the same and choosing the fuel element diameter and inner clad diameter from within their tolerances such that the gap conductance becomes minimum. For that  $d_p = 14.23 \text{ mm}$  and  $d_i = 14.4894 \text{ mm}$ . Applying swelling and thermal expansion model to these respectively,

$$d_p=14.3723 \text{ mm, and } d_i=14.52 \text{ mm}.$$

So the worst case value of gap heat transfer coefficient at swelling of 3% is given by eq.(2.13) as,

$$h=680.52 \text{ Wm}^{-2}\text{K}^{-1}.$$

This value is about 12% of the nominal gap conductance and shows that gap conductance has a very wide distribution. This results from sharp dependence of gap conductance on inner clad diameter and fuel element diameter through a logarithmic function. It should be mentioned here that there are chances of an overlap or interference of fuel element and clad inner surface at the tolerance limits at other extreme and this makes the largest gap conductance theoretically infinite. The detailed analysis, about the distribution of gap conductance and the fraction of area for which pellet and clad are in contact, has been presented in the next chapter.

#### (c) Inner Clad Surface Temperature

Choosing the worst case data values of inner clad diameter and outer clad diameter from within the tolerance limits, that is,  $d_i=14.3106 \text{ mm}$  and  $d_c=15.26 \text{ mm}$ , and applying thermal expansion models to these,  $d_i=14.341 \text{ mm}$ , and  $d_c=15.292 \text{ mm}$ .

Using these values in eq.(2.16), the worst case value of inner clad surface temperature is given as,

$$T_s=608.2 \text{ K}$$

This is about 1% higher than the nominal value of inner surface temperature given earlier, showing that the inner clad surface temperature variation is rather small.

#### (d) Fuel Center-line Temperature

For a swelling of 3%, for the worst case fuel pellet diameter should be minimum, inner clad diameter and outer clad diameter should be maximum within their tolerance limits. This can be checked by differentiating the appropriate equation for fuel center-line temperature. For a fuel swelling of 3% (including thermal expansion) and using thermal expansion of the cladding as

given in Chapter 2 and choosing other parameters appropriately, that is,

$$d_p = 14.3723 \text{ mm}, d_i = 14.52 \text{ mm}, \text{ and } d_c = 15.2925 \text{ mm}$$

and using other constants, eq.(2.18) gives the worst case fuel centre-line temperature  $T_{cl}$ , as,

$$T_{cl} = 3802.28 \text{ K}.$$

This is about 70% higher than the nominal value of fuel center-line temperature which indicates that the spread in fuel center-line temperature is considerable. Also, this temperature is higher than the melting point temperature of the fuel and therefore in a few cases the fuel rods do melt near the centre. In the next chapter detailed analysis of the distribution of fuel center-line temperature and of the probability of fuel melting, are presented.

#### (e) Fuel Rod Internal Pressure

The internal pressure is directly dependent on the volume available for fission gases released from the fuel elements. For the worst combination, including swelling=3%, inner clad diameter and clad length minimum, mass of fuel maximum and density of fuel minimum, there is no volume left for released fission gases. However, it is unlikely that mass and density of a fuel element can be varied independently to a large extent (so that maximum mass and minimum density occurring simultaneously in a fuel element is ruled out). To take this into account, nominal values of fuel mass and fuel density are considered for obtaining the worst case value of fuel rod internal pressure. For worst case values of inner clad diameter and clad length and swelling=3%, the volume available for fission gases from eq.(2.22) is,

$$V = 6.48 \times 10^{-7} \text{ m}^3.$$

From eq.(2.23), number of moles of fission gases ( $n$ ) is,

$$n = 1.1946 \times 10^{-3} \text{ mol}.$$

The eq.(2.21), therefore gives fuel rod internal pressure (p) as,

$$p=90.3 \text{ bar.}$$

This pressure is a little below the external coolant pressure and about 2.5 times the nominal value of fission gas pressure as obtained earlier. So the pressure spread is also quite wide. Again, the detailed analysis of fission gas internal pressure and its comparison with external coolant pressure is presented in the next chapter.

The results of deterministic and worst case analysis are tabulated in Table 3.2.

TABLE 3.2 RESULTS OF NOMINAL AND WORST CASE DETERMINISTIC ANALYSIS

	Nominal values	Worst case values
Fractional fission gas release	11.4 %	12.08 %
Gap heat transfer coefficient	$5581.3 \text{ Wm}^{-2} \text{ K}^{-1}$	$680.52 \text{ Wm}^{-2} \text{ K}^{-1}$
Inner clad surface temperature	602.72 K	608.2 K
Fuel center-line temperature	2232.2 K	3802.28 K
Fuel rod internal pressure	36.65 bar	90.3 bar

## CHAPTER 4

### PROBABILISTIC ANALYSIS

#### 4.1 INTRODUCTION

The common practice in the fuel rod analysis or design is to use values at tolerance limits to obtain the various performance parameters. For each calculation, a combination of unfavourable tolerances is selected giving a so called worst case dataset(chapter 3). Parameter distributions are not considered in this procedure.

The results of the worst case calculations are very conservative but the degree of conservatism is difficult to quantify. Therefore probabilistic calculations based on distributions of as-fabricated data would be helpful for fuel rod design. This is achieved by the following three steps [12]

Step 1: Statistical distributions for the important input parameters appearing in the intermediate or output performance parameter models are determined.

Step 2:Developing/deriving, based on appropriate simplifications, of intermediate parameter model and output parameter model which determine the fuel rod behaviour.

Step 3: Monte Carlo calculations with the above models on the basis of distribution of input parameters are performed to produce the probability distribution of output parameters. From these distributions various deductions about fuel rod behavior are determined.

#### 4.2 STATISTICAL DISTRIBUTIONS OF INPUT PARAMETERS

The important as-fabricated parameters are pellet density,

pellet mass in each fuel bundle/pin, pellet diameter, inner clad diameter, outer clad diameter, average fuel grain radius and length of the clad or sheath. The nominal value and tolerances of all the above parameters are provided in the AERB/NPC datasheet[11]. In many manufacturing processes, the most frequently encountered distribution of dimensions is the normal distribution so in the absence of specific information regarding the same, a normal distribution is assumed for all the above parameters which is in unison with the nominal values and the tolerances as given in [11].

From the known nominal values and tolerance limits, and assuming normal distribution, the actual statistical distribution of the input parameters is obtained. This uses standard input of a probability density curve of a normal distribution through a input file. This method of generating statistical distribution of input parameters facilitates easy incorporation of any other distribution function for any of the input parameters (say, binomial distribution, Poisson distribution, Maxwellian distribution or Gamma distribution, etc.). In conducting the Monte Carlo simulations to estimate the distribution of the output parameters, the uncertainty in each input parameter was modeled as a normal distribution with the nominal value being the mean and one-sixth of the tolerance being the corresponding standard deviation.

In addition to the fabrication related parameters two other groups of parameters determine the behaviour of the fuel rods and are inputs to the overall modeling process ;

---- modelling parameters ;

---- data related to the power history of the fuel rod.

The former include the fractional fission gas release, fuel swelling (including fuel thermal expansion ) and thermal expansion

of the cladding. From the mathematical models chosen/developed for these parameters, their distributions are obtained depending upon the distribution of the related as-fabricated parameter or other constants. As mentioned earlier, for fuel swelling, a discrete set of values is used in the present work. Other modeling parameters like cladding creep, structural changes, stressing etc. have been neglected in view of their low importance as explained in Chapter 2.

Power history data are considered to be fixed for one analysis. These include linear thermal power rating and burn-up. To analyze the behaviour of the fuel rods in a complete core, the fuel rods can be grouped, each group being characterised by one power history. Then the statistical analysis can be performed for each group and finally the results for all groups can be superimposed in order to get the distributions for the whole core.

In addition there are certain parameters mostly dependent on the material properties like thermal conductivities or thermal expansion coefficients etc. These have been assigned a fixed value dependent on the operating conditions. Certain output parameters also enter as input to certain models (for example fission gas release model needs fuel center-line temperature and fuel surface temperature). In such cases values have been assigned for the required input parameters and it has been verified that these assumed values are not drastically different from the output values.

The calculation of fractional fission gas release and fuel rod internal pressure requires the residence time of fuel element in the reactor core. This has been fixed for the fuel element on the basis of power history data, viz. a constant burnup of the fuel with the assigned linear thermal power rating. Further, the

coolant conditions are assumed to be constant like the mass flow rate and uniform flow of the coolant. This resulted in a fixed value for the average coolant temperature and for clad to coolant heat transfer coefficient.

#### 4.3 INTERMEDIATE AND OUTPUT PARAMETER MODELS

This part has been generally covered under Chapter 2. Certain simplifying assumptions have been made to arrive at these models.

These have been listed below: (1) For fission gas release model, a spherical shape for  $\text{UO}_2$  grains has been assumed. Then diffusion model based on Fick's law has been applied to find the fractional fission gas release. (2) Other less significant factors contributing to fission gas release have been neglected like bubble formation and bubble growth.

(3) For gap conductance, a one-dimensional heat transfer model has been assumed across the gap.

(4) Also the thermal conductivity of the gas mixture has been assumed to be constant based on a definite composition of prefilled gas and released fission gases.

(5) For temperature drops across the pellet, clad, gap and from the clad to coolant one-dimensional thermal conduction and convection models have been assumed. Axial conduction has been neglected and any azimuthal variation in the temperature has been ignored.

(6) Average values have been assigned to certain associated variables for example the coolant temperature, heat transfer coefficient between coolant and the clad, and thermal conductivity values.

(7) For fission gas pressure calculations ideal gas law has been used for the fission gases.

(8) Some variables depend upon certain output parameters. For example, the gap temperature enters into fission gas pressure calculation. In such cases iterations have been neglected and a reasonable average value based on preliminary calculations has been assumed. Nevertheless, it has been assured that the final circumstances are not very different from those assumed in their calculation.

It should also be mentioned that the nominal values of the outputs of the models have generally been compared with the operating values for PHWRs, where possible, and are found to be not very different from these values. Thus, the models give a fair idea of the effect of various input parameters and their distributions on the variation of output parameters of interest, inspite of the simplifying assumptions mentioned above.

#### 4.4 MONTE CARLO CALCULATIONS OF THE OUTPUT AND OTHER PARAMETERS

On the basis of the distributions of as-fabricated parameters and modeling parameters as described in Section 4.2, simulations of the fuel rod behaviour can begin in the following way :

- For each (variable) input parameter of the fuel rod model, select one value randomly corresponding to the distribution which is valid for this parameter.
- Combine the random dataset with the other input parameters and perform one calculation of the output parameter or intermediate parameter with the mathematical models already given.
- Store the output values of interest for all the output parameters (These include the fractional fission gas release, volume available for the fission gases, gap heat transfer coefficient, inner clad surface temperature, fuel center-line temperature and fission gas pressure in the fuel rod ).

- Repeat this procedure many times. (Pilot runs showed that 80,000 trials would provide smooth estimation of the probability distributions. )
- Generate a frequency distribution of the output variable of interest which describes the status of the group of fuel rods under consideration.

To get a satisfactory accuracy of the frequency distribution the number of trials required for one analysis is generally above 50,000; 80,000 trials were found to give a reasonably smooth distribution in all cases. For plotting the frequency distributions, the total range of the variables/parameters of interest was divided into 50 equal intervals.

#### 4.5 RESULTS

The resulting distributions of all the output parameters of interest are shown in Figs.1 to 12. These results have been generated on the basis of distributions and models which are to some extent preliminary. This is because the input distributions are based on the tolerance values alone, and as mentioned in Section 4.3, certain simplifying assumptions have been made to simplify the mathematical models used. Nevertheless, in the following the results are discussed and compared with those of the deterministic calculations as given in Chapter 3.

##### (a) Fractional Fission Gas Release

Figures 1 and 2 show the distributions of fractional fission gas release with 80,000 trials and average fuel grain radius  $50^{+3}_{-3} \mu\text{m}$  and  $30^{+3}_{-3} \mu\text{m}$ , respectively. The distributions in both the cases are quite symmetric about the nominal value. The nominal value is 11.36% for average fuel grain radius of  $50 \mu\text{m}$  and is 17.6% for average fuel grain radius of  $30 \mu\text{m}$ .

The worst case analysis at 50 $\mu$ m fuel grain radius shows that fractional fission gas release is 12.08%. From our distribution 99.91% of the values are below this value which are more favourable than the worst case value (because low release of fission gas is desired to keep fission gas pressure low as well as to keep temperature drop across the gap low). So the degree of conservatism in the worst case analysis is 99.91%.

(b) Gap Heat Transfer Coefficient or Gap Conductance

Figures 3 to 5 show the distributions of gap heat transfer coefficient with 80,000 trials for fuel swelling of 2%, 3%, and 4% respectively. The distributions in these cases are rather asymmetric and close to Gamma or Erlang distribution functions. Also the most probable (mode) and average values of gap conductance are significantly different from each other.

The reason for this is that there is a possibility of interference of clad and fuel pellet (Fig.6) and this increases as swelling increases. Whenever an interference occurs the gap conductance is assigned a large value. This introduces the asymmetry in the observed distribution for gap conductance, which shows an increase with the swelling. This also explains the observation that the difference between average value and most probable value is more for higher swelling because the chances of interference increase. In fact for a swelling of ~3.5% the nominal values of pellet diameter and inner clad diameter come into conflict and the nominal value of gap conductance attains a very large value >> most probable value. A higher swelling gives a higher value of gap conductance which is desirable but it also increases the interference of pellet and clad which is undesirable. Among the various factors affecting the gap conductance, fuel swelling, inner clad diameter, pellet diameter,

and thermal expansion of the cladding play an important role. Fractional fission gas release appears only indirectly by influencing the thermal conductivity of the gap. The comparison with the worst case dataset shows that all the values of gap conductance are above the worst case value<sub>A</sub> <sup>(See Table 3.2)</sup> and hence are more favourable. So the degree of conservatism is nearly 100% !

#### (c) Inner Clad Surface Temperature

Figure 7 shows the distribution of inner clad surface temperature with 80,000 trials. It is independent of fuel swelling because swelling affects the pellet diameter and the gap which do not come into picture in calculating inner clad temperature. The distribution is rather symmetrical about the nominal value and is close to normal distribution function (which means that the average value is very close to the most probable value). Among other modeling parameters fractional fission gas release also does not affect the clad temperature but thermal expansion of the cladding does affect it. Among as-fabricated parameters, only the clad inner and outer diameters affect the clad temperature.

The worst case analysis shows that the maximum inner clad surface temperature is 608.2K. From our distribution, Fig. 7, 99.89% values are lower than this value which are more favorable (since least inner clad surface temperature is desired). So the degree of conservatism is 99.89%.

#### (d) Fuel Centre-line Temperature

Figures 8 and 9 show the distributions of fuel center-line temperature with 80,000 trials for swelling of 2% and 3% respectively. At 4% swelling, the gap between pellet and clad vanishes for more than 80% area of inner clad surface. Therefore for major fraction of the area (>80%) the gap conductance becomes very high and fuel centre-line temperature close to minimum. These

very large number of values at minimum overshadow any variation of fuel center-line temperature due to various parameters. This trend is also clear from fig.9 at 3% swelling. Further, the distribution is asymmetric about most probable value and is closer to Gamma or Erlang distribution function ( $x^n e^{-x}$ ). In addition nominal value is quite different from most probable value of the fuel center-line temperature.

The reason for asymmetric distribution is that there is a significant interference of clad and pellet. For all these contact values, temperature drop across the gap becomes very small. These combinations of pellet and clad diameters are assigned a fixed large value of gap conductance. This also leads to a big difference between average and most probable value (because the former is independent of the interference of clad and pellet and the latter is affected by the percentage of contact). This reasoning is strengthened by the observation that at 1% swelling (when interference is very small) the distribution of fuel center-line temperature is close to normal distribution and average and most probable values almost coincide. Increase in swelling makes the distribution more and more asymmetrical. Large swelling increases the pellet-clad interaction but reduces fuel melting (this is because the temperature drop across the fuel to clad gap reduces a lot as swelling increases, reducing the fuel centre-line temperature).

Among the modeling parameters affecting fuel center-line temperature, fuel swelling and thermal expansion of the cladding have a marked effect. Fractional fission gas release has only an indirect effect through gap conductivity by altering composition of gases in the gap. As-fabricated parameters like pellet diameter, inner clad diameter and outer clad diameter also affect it. In the

power history parameters, it is directly related to linear thermal power rating.

The worst case analysis at 3% swelling gives a value of 3802.3K for fuel center-line temperature. From the distribution at 3% swelling, 99.89% values are below this value. So the degree of conservatism is 99.89%.

(e) Fission Gas Pressure in Fuel Rod

Figures 10 to 12 show the fission gas pressure distributions with 80,000 trials for swelling 2%, 3% and 4% respectively. The distributions are asymmetric about the most probable values and are closer to Gamma or Erlang distribution function. Also average value is somewhat different from most probable value in the fuel rod pressure distributions.

The reason for this lies in the volume available for fission gases released from the fuel pellets. It is obtained as the difference of total volume available inside the cladding or sheath and volume occupied by  $\text{UO}_2$  fuel pellets. This available volume has a large variation and for the worst case combination of inner clad volume and material volume, this vanishes completely. These cases are taken care of by assigning a large value to the internal pressure (~350 bar). This introduces the asymmetry observed in the distribution functions. This is also evident from the increasing asymmetry in the distributions with increase in swelling. At 2% swelling, the distribution has an average value almost coincident with the most probable value and at 4% swelling the two are farthest from each other. Around 4.7% swelling, the average value of volume available for fission gases vanishes completely and a dangerous condition of large internal pressure can arise. So a large swelling is undesirable from fission gas pressure point of view.

Among the modeling parameters affecting the fission gas pressure, fractional fission gas release, swelling as well as cladding thermal expansion all play a role. Among fabrication parameters, inner clad diameter, pellet density, pellet mass, and clad length become important. Fuel burn-up and linear thermal power rating are equally important in determining the fission gas internal pressure.

The worst case analysis with fuel mass and density independent (that is, maximum mass of the pellets with minimum density, occurring simultaneously) leaves no volume for the fission gases in the fuel pin. ternal pressure. However, if the dependence of fuel mass and fuel density is considered (that is mass is taken as proportional to the density ), the worst case pressure becomes 90.3 bar (see Table 3.2). The distribution of fission gas pressure in fig.11 gives the degree of conservatism in the worst case analysis to be 93.17%.

In the next chapter, we present a statistical parametric study involving the input variables and the fuel element behavior (gap conductance, central line temperature and fission gas pressure ).

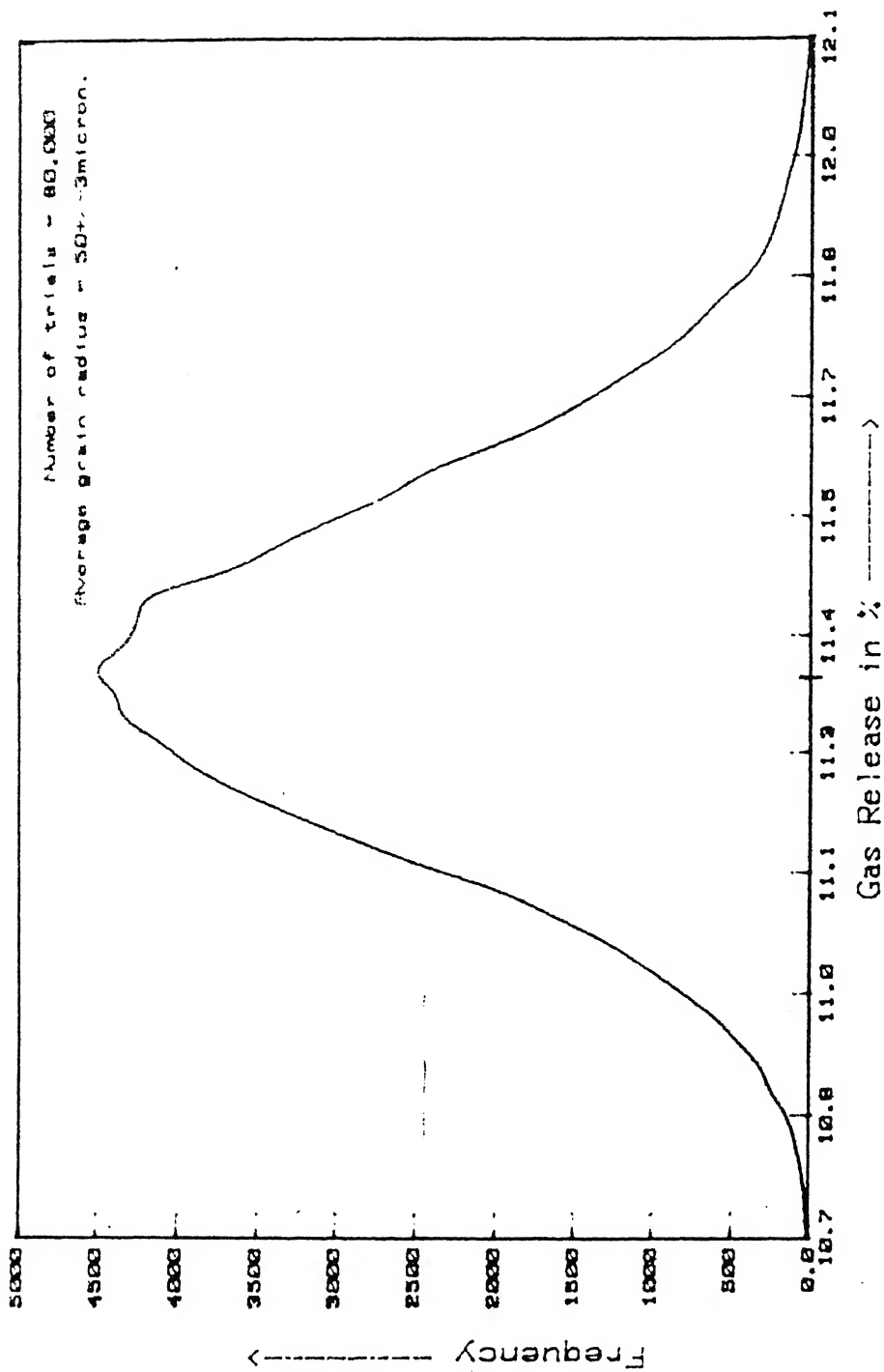


Fig. 1 FISSION GAS RELEASE DISTRIBUTION

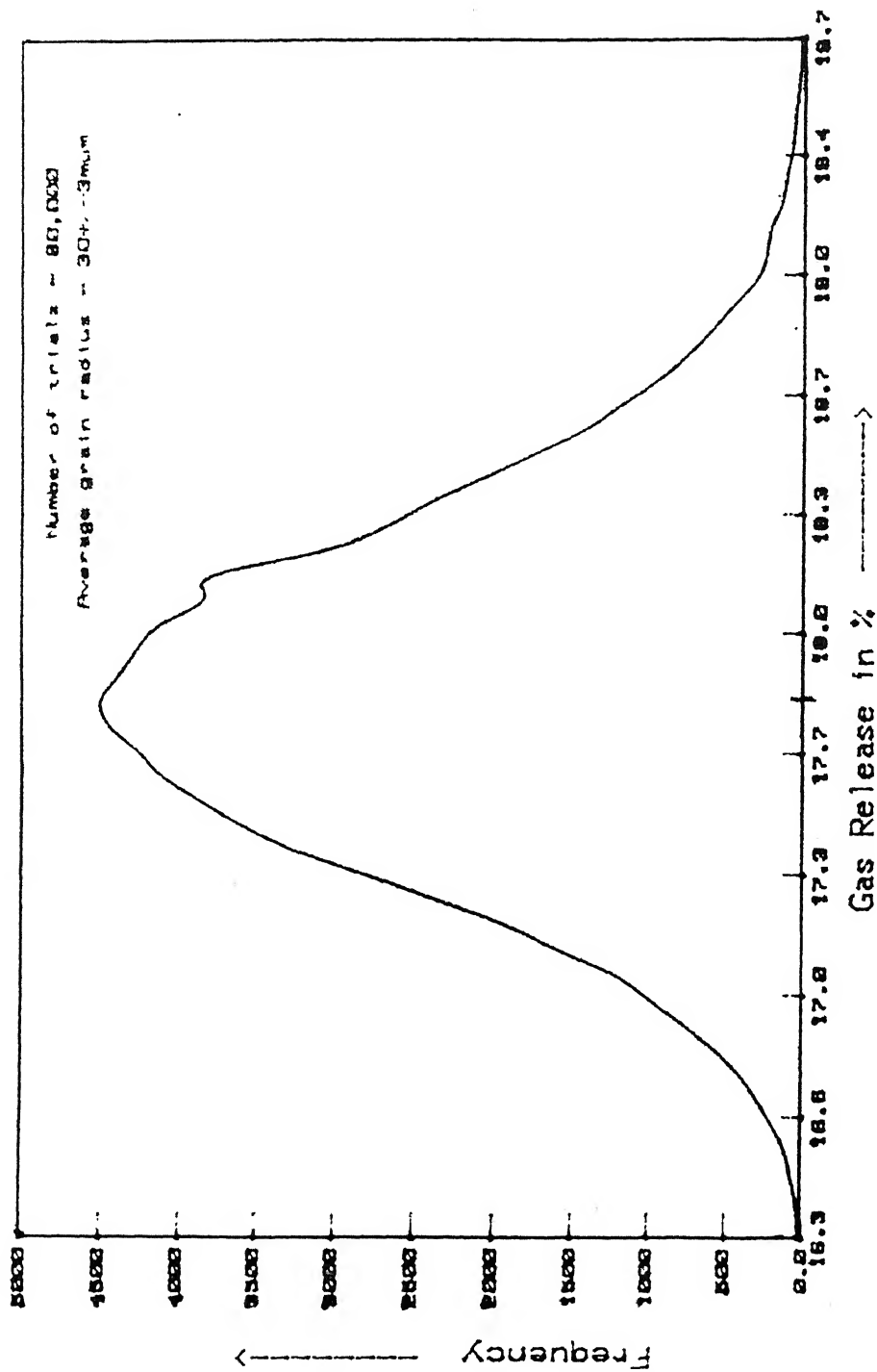


Fig. 2 FISSION GAS RELEASE DISTRIBUTION

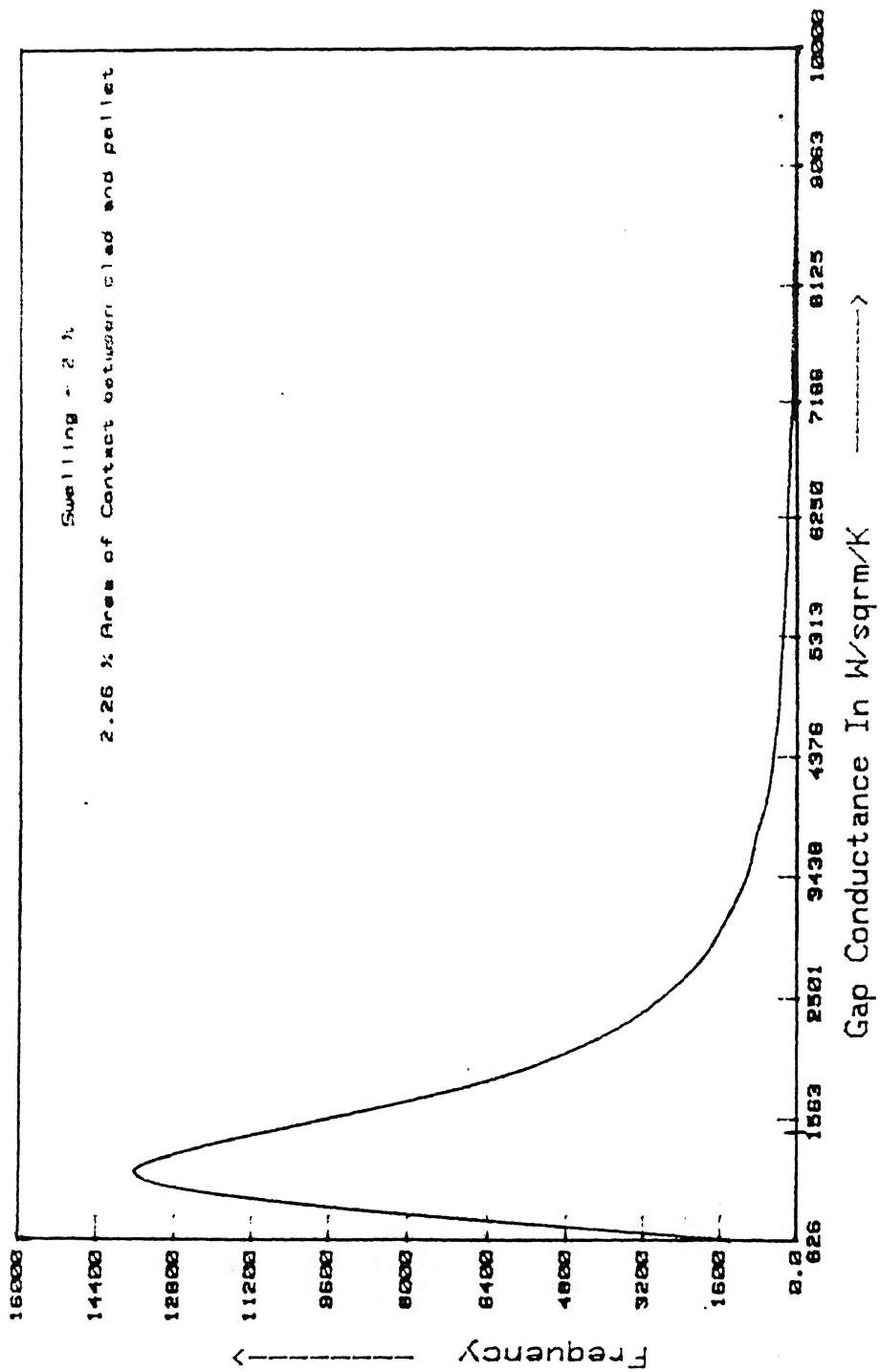


Fig. 3 GAP CONDUCTANCE DISTRIBUTION

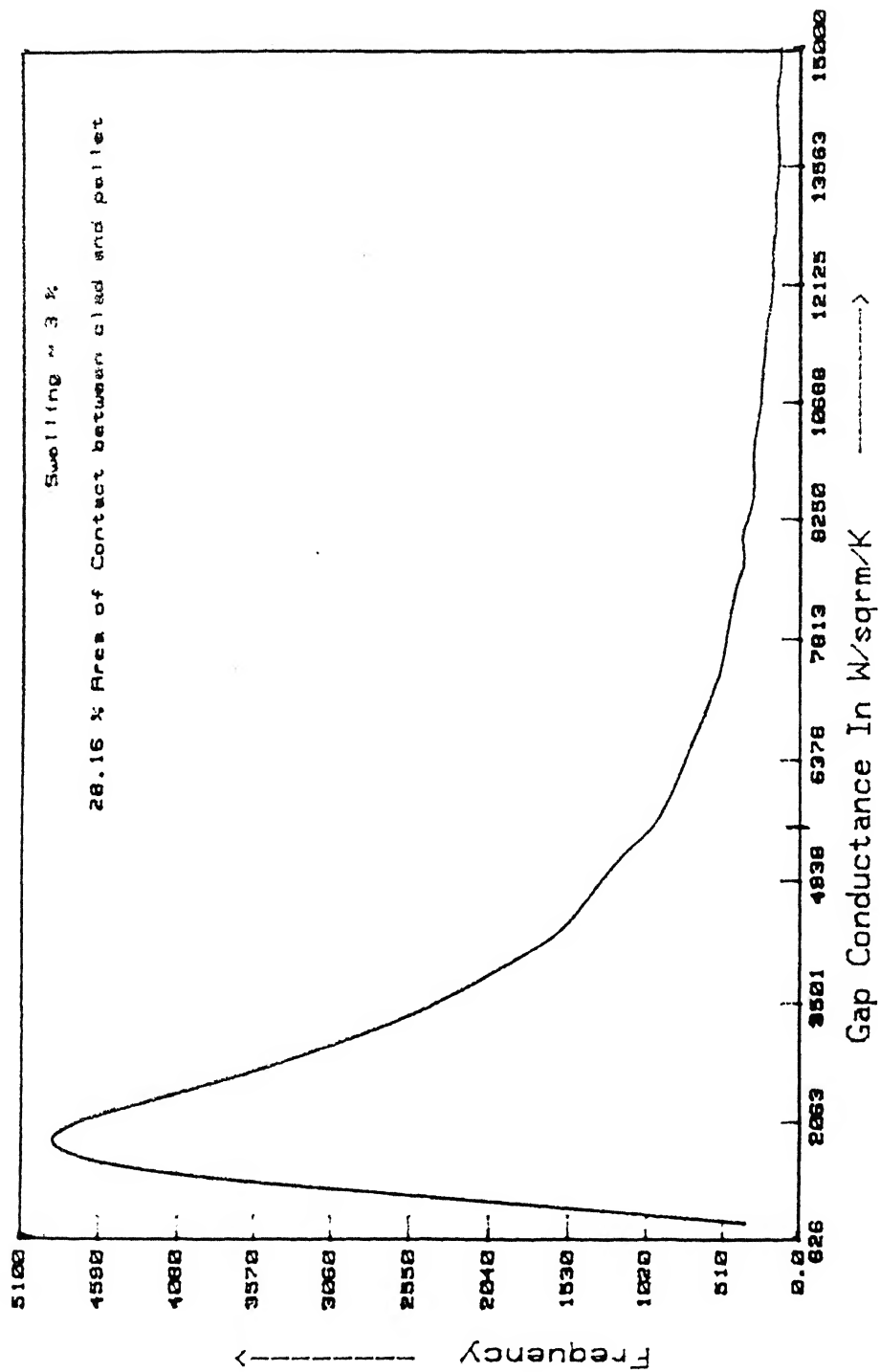


Fig. 4 GAP CONDUCTANCE DISTRIBUTION

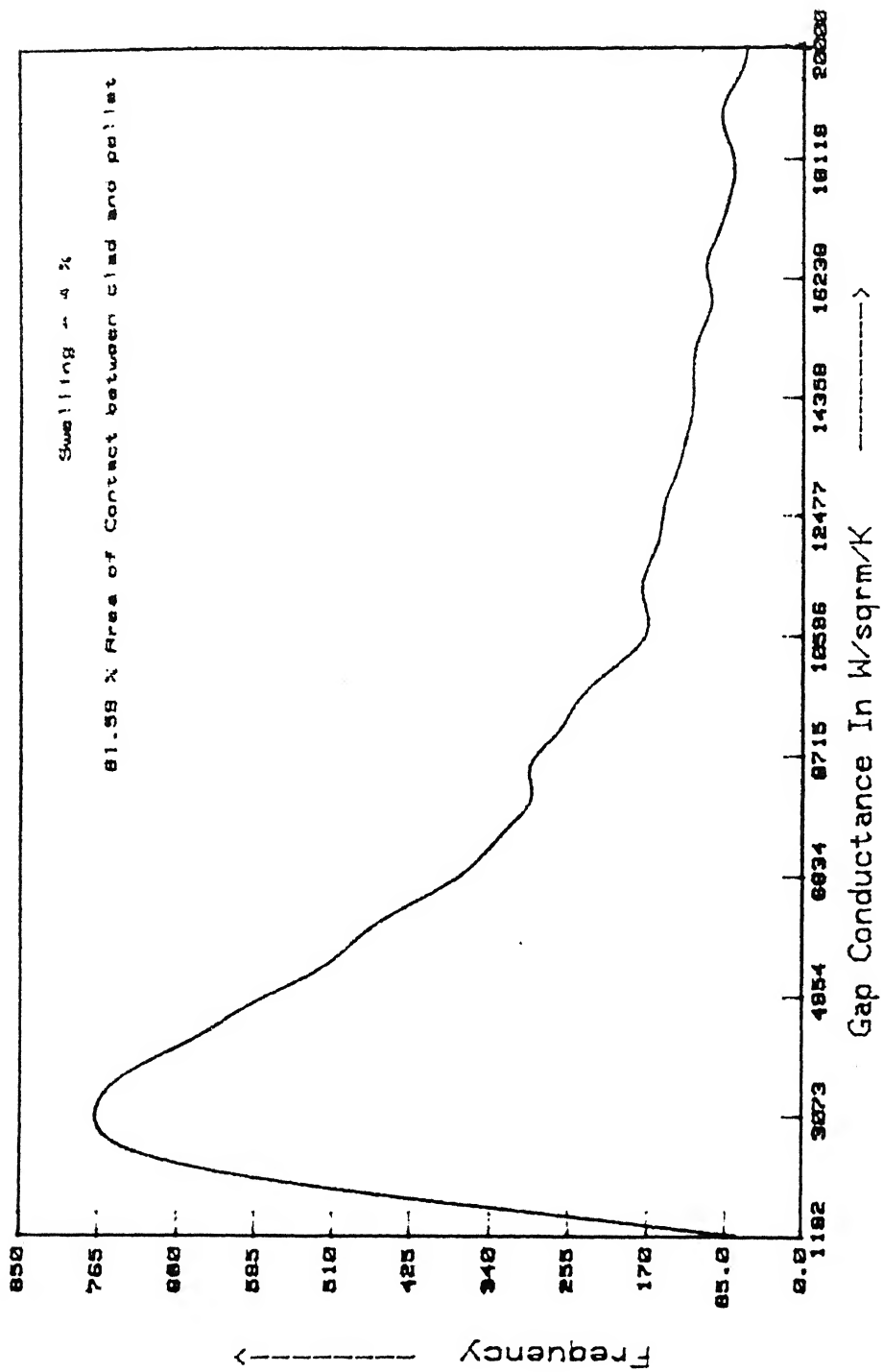


Fig. 5 GAP CONDUCTANCE DISTRIBUTION

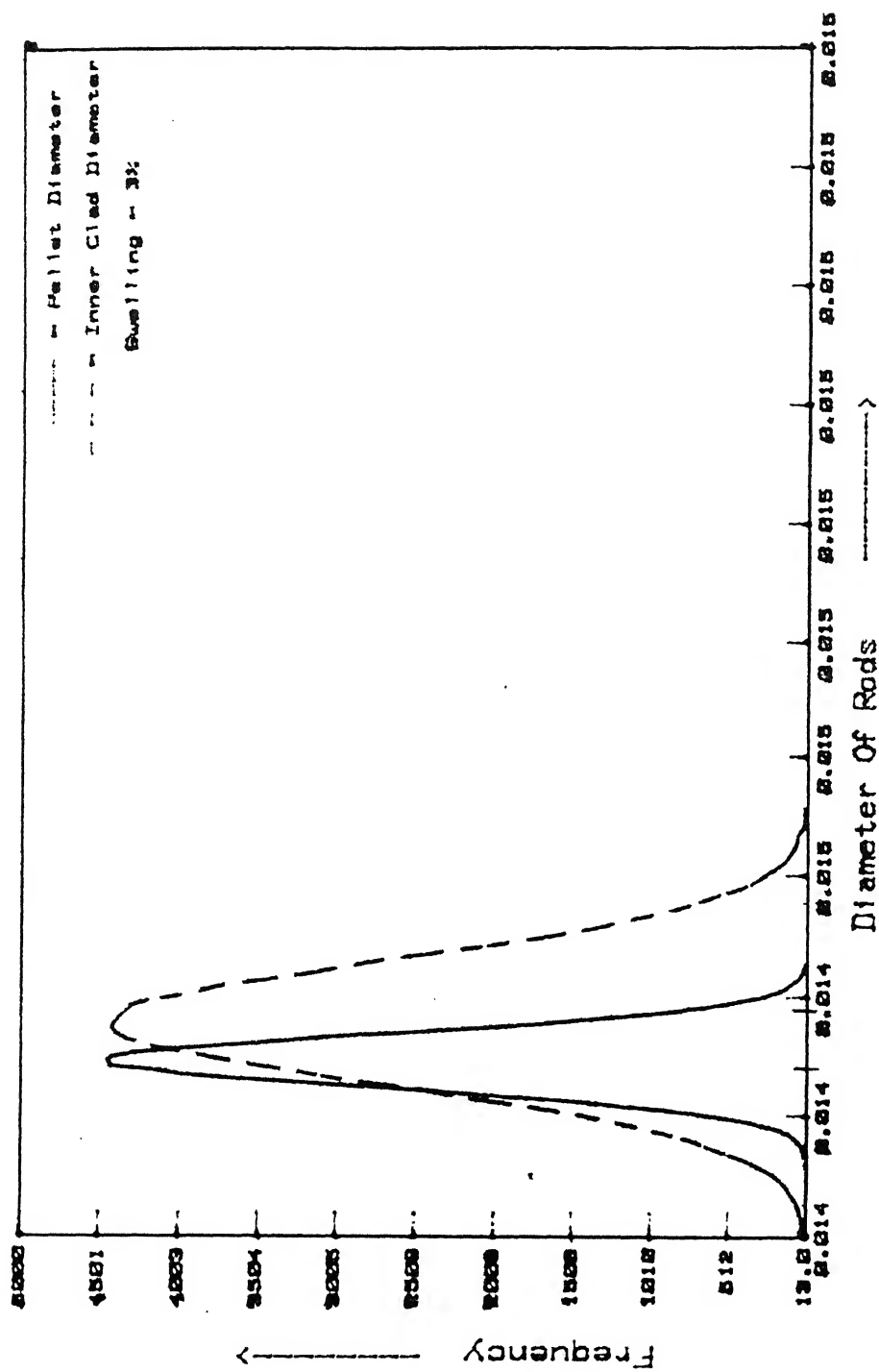


Fig. 6 PELLET-CLAD INTERFERENCE DISTRIBUTION

[ The standard deviation of both above distributions are assumed to be one-sixth of the respective tolerances for  $d_p$  and  $d_i$  shown in Table 3.1 ]

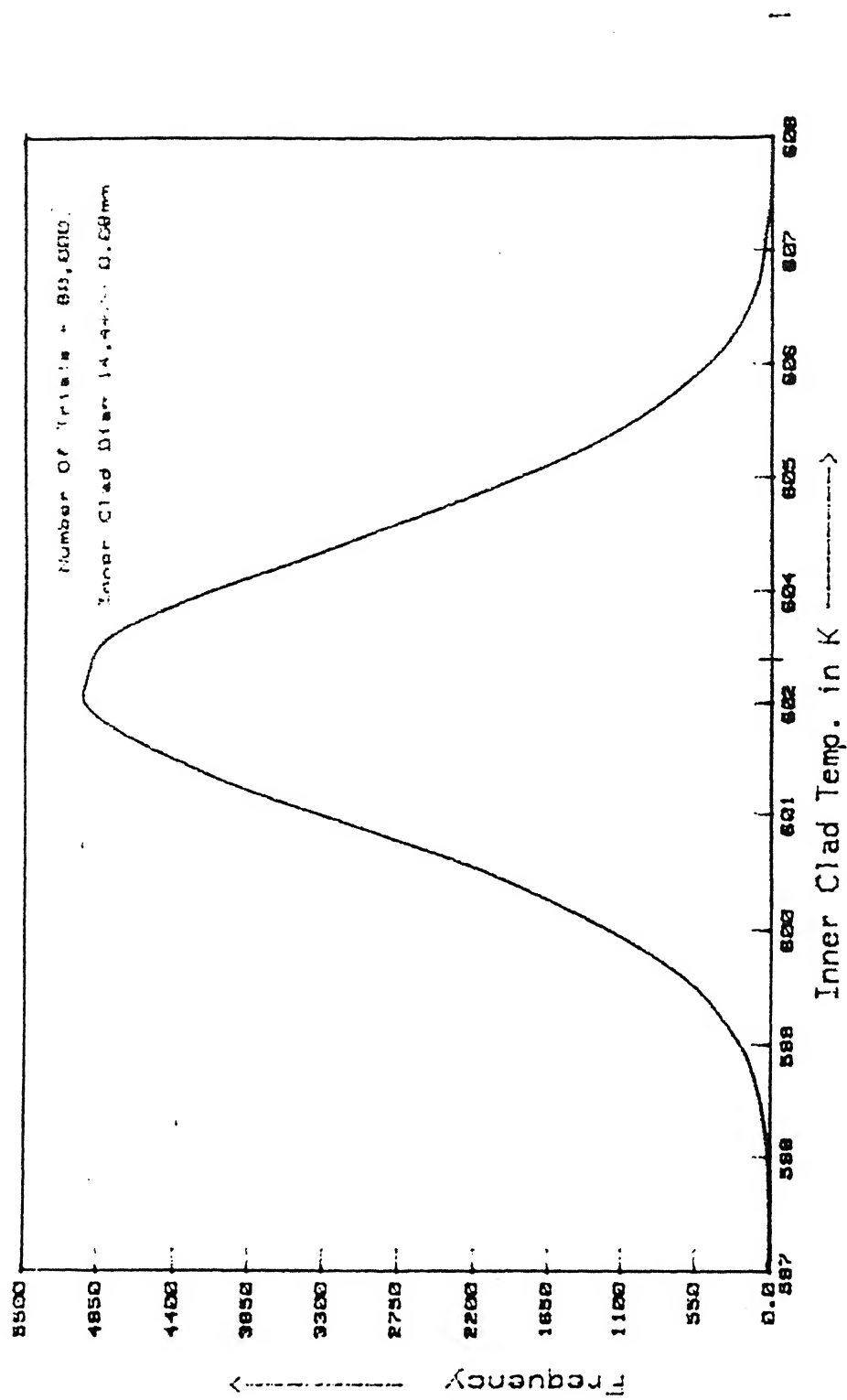


Fig. 7 INNER CLAD SURFACE TEMPERATURE DISTRIBUTION

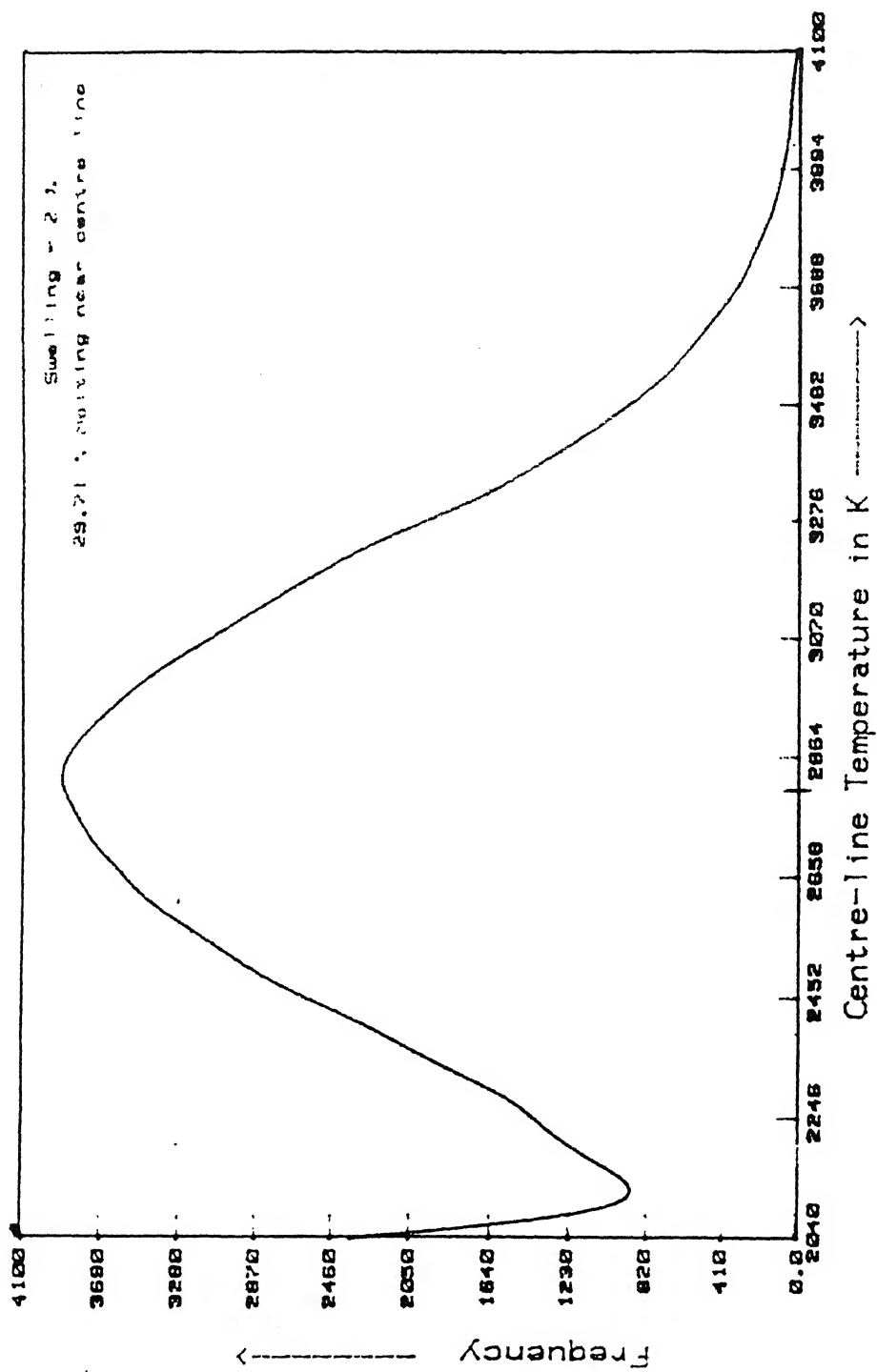


Fig. 8 FUEL CENTRE-LINE TEMPERATURE DISTRIBUTION

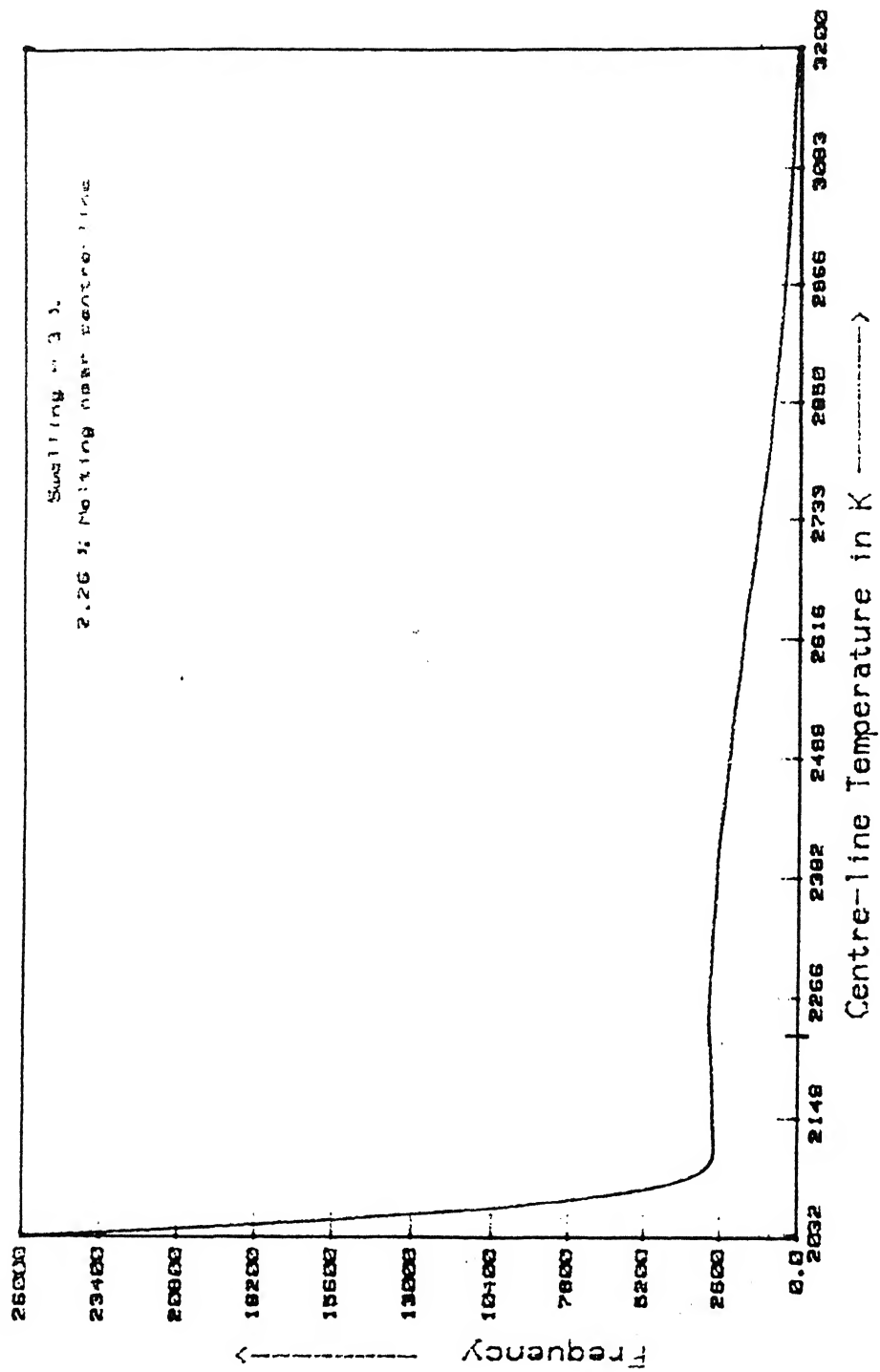


Fig. 9 FUEL CENTRE-LINE TEMPERATURE DISTRIBUTION

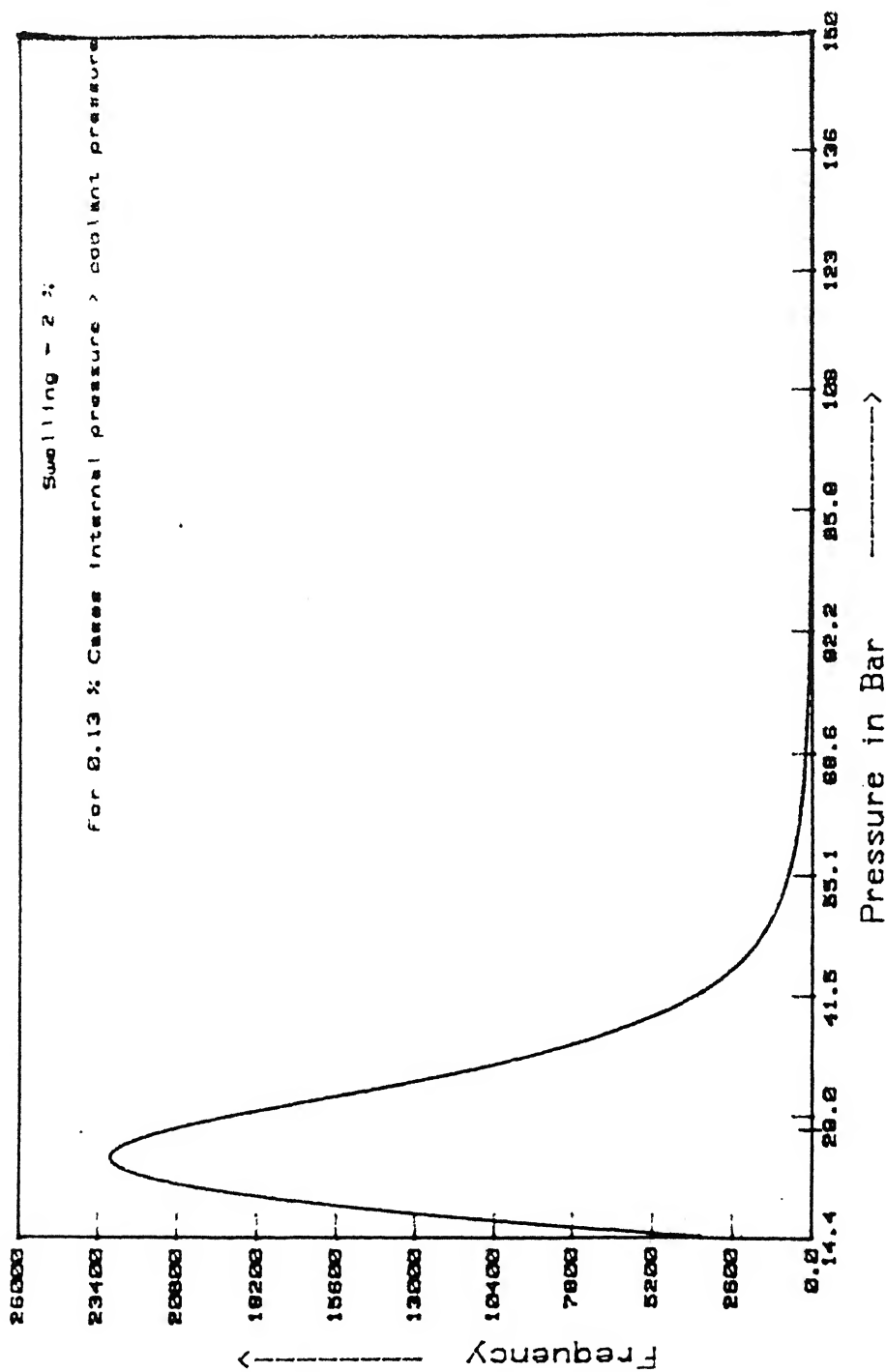


Fig. 10 FISSION GAS PRESSURE DISTRIBUTION

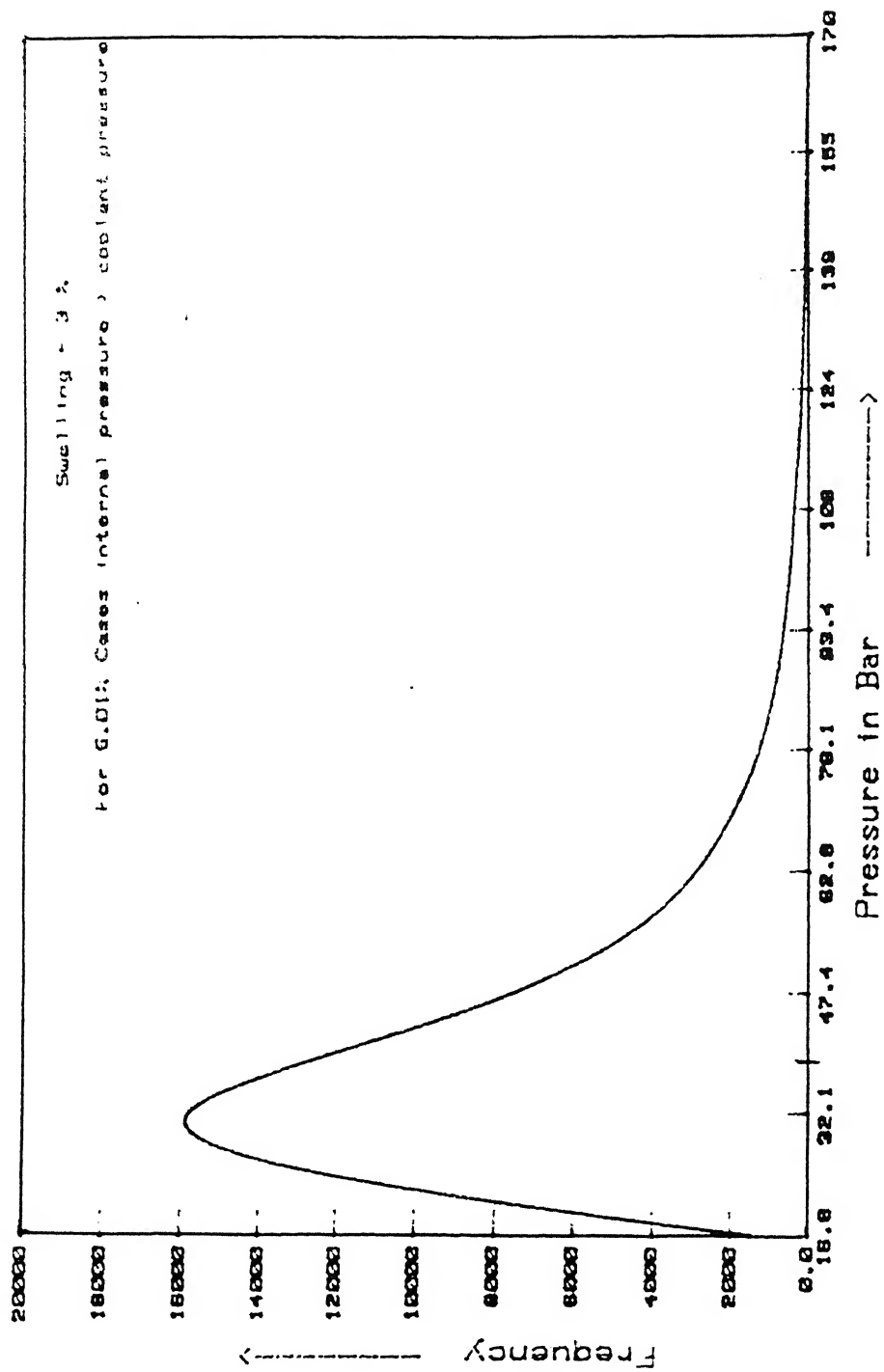


Fig. 11 FISSION GAS PRESSURE DISTRIBUTION

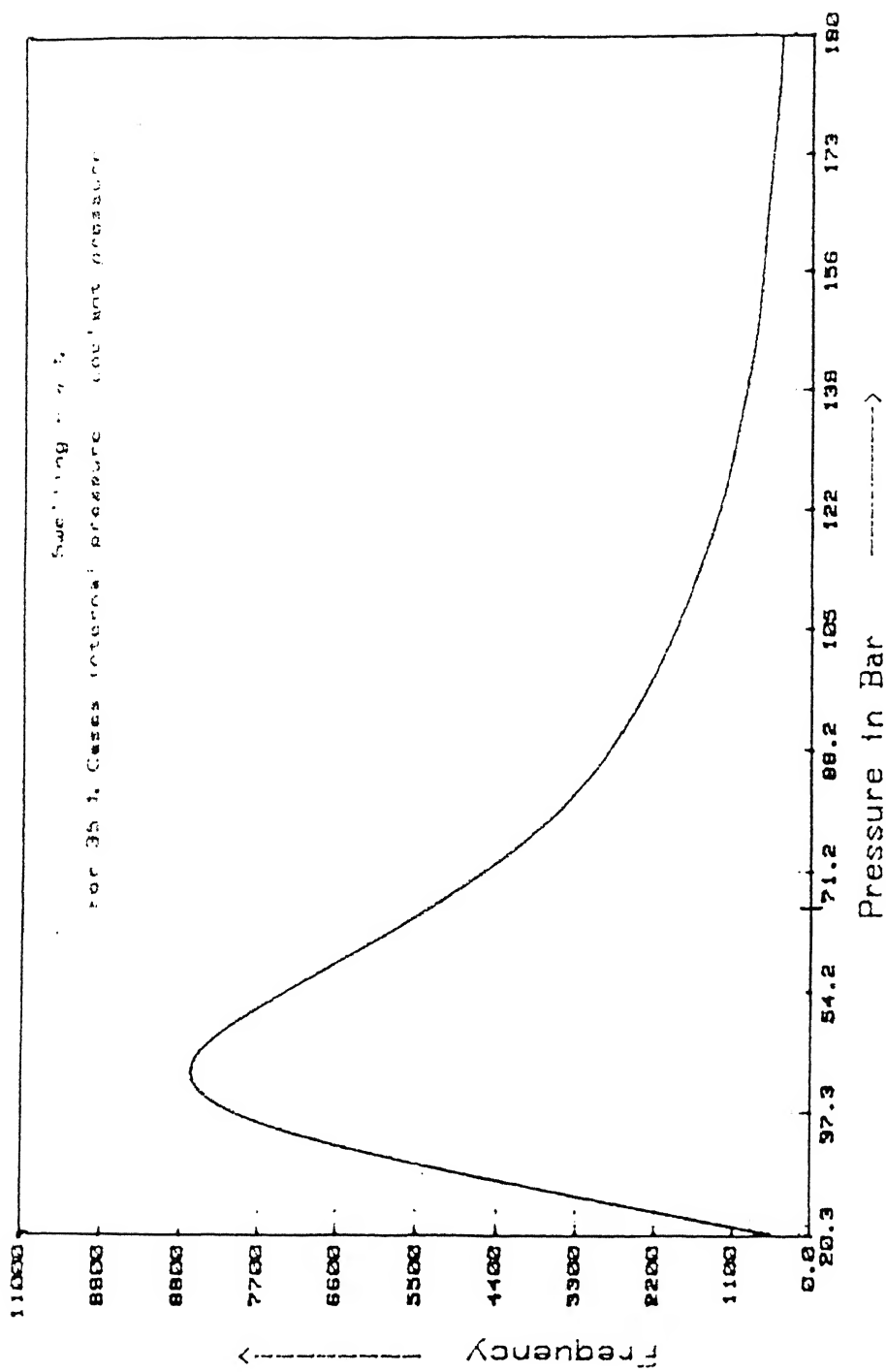


Fig. 12 FISSION GAS PRESSURE DISTRIBUTION

## CHAPTER 5

### DESIGN EVALUATION USING ORTHOGONAL ARRAYS

#### 5.1 PERFORMANCES OF INTEREST

Various output parameters of interest which have been considered in the present work are gap conductance, fuel center-line temperature, fractional fission gas release, clad maximum temperature and fission gas internal pressure. Out of these the gap conductance( $h_g$ ), fuel center-line temperature( $T_c$ ) and fission gas internal pressure( $p$ ) are relatively more important from design point of view and have been analysed using a statistical experimental design framework in this chapter.

As given in Chapter 2, the governing equations (models) for these variables are as follows:

$$h_g = \frac{2 k_g}{d_p \ln(d_i/d_p)}$$

$$T_c = T_{cool} + \frac{q'}{\pi} \left[ \frac{1}{4k_p} + \frac{1}{d_p h_g} + \frac{\ln(d_c/d_i)}{2k_c} + \frac{1}{d_c h_c} \right]$$

$$T_i = T_{cool} + \frac{q'}{\pi} \left[ \frac{1}{d_c h_c} + \frac{\ln(d_c/d_p)}{2k_c} \right]$$

$$p = n f R T_g / V$$

where

$k_p$ ,  $k_c$  and  $k_g$  are thermal conductivities of fuel pellet, cladding and fuel to clad gap respectively,

$h_c$  is the heat transfer coefficient from clad to coolant,

$q'$  is linear thermal power rating,

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$d_p$ ,  $d_i$  and  $d_c$  are diameters of fuel pellet, inner clad and outer clad surface respectively.

$T_{cool}$  is the average coolant temperature,

$n$  is the number of moles of fission gases produced,

$f$  is fractional fission gas release from the fuel which is a function of average fuel grain radius  $a$ ,

$V$  is the volume available for fission gases,

$T_g$  is the mean gap temperature, and

$R$  is universal ideal gas constant.

The gap conductance ( $h_g$ ) is dependent on two input parameters which are pellet diameter ( $d_p$ ) and inner clad diameter ( $d_i$ ). The fuel center-line temperature ( $T_c$ ) depends on three input parameters, including outer clad diameter ( $d_c$ ) in addition to the above two. Inner clad diameter and outer clad diameter affect the clad maximum temperature ( $T_i$ ). The fission gas internal pressure ( $p$ ) depends on average fuel grain radius through fractional fission gas release ( $f$ ) and on the volume available for fission gases which is the difference of volume enclosed by the clad and volume of fuel residing inside the clad.

## 5.2 (a) EXPERIMENTAL PLAN FOR STUDY OF $h$ , $T_c$ , and $T_i$

The purpose of the first set of designed experiments was to determine which of the input factors are important in the fuel manufacturing process and to find the optimum set of parameter values ( $d_p$ ,  $d_i$ ,  $d_c$ ), i.e., the best levels of those factors. The method of orthogonal arrays is the basis of these experiments.

The actual runs led to Monte-Carlo simulation with 5000 trials within the parameter tolerance levels as given in Chapter 3. In doing the Monte Carlo simulations for the designed

experiments, each tolerance range given in Table 3.1 was assumed to be  $12\sigma$ , reflecting a  $c_{pk}$  ( process capability index [12.p290] ) of 2.0, assumed to be applicable also for the AERB/NPC manufacturing data given in [11].

Considering three factors affecting the output parameters and taking two levels of each, a "full factorial" (eight-point or L-8) experiment [14] is run and this is done for evaluating the nominal and average values and for standard deviation under "noise" of each of the performance or output parameters ( $h_g$ ,  $T_c$  and  $T_r$  ). The experimental design is shown in Table 5.2.1.

The data may be analysed in two ways. First, the response of nominal and average values of the output can be taken up. That tells which factors have a major influence on the *location* ( average response) of the process. In the second analysis, the estimated *variance* at each of the eight data sets ( experimental conditions ) are taken as performance characteristics.

Table 5.2.1 L-8 Orthogonal array for 3 factor at 2 level design

Run #	A	B	C	output
1	-1	-1	-1	
2	+1	-1	-1	
3	-1	+1	-1	
4	+1	+1	-1	
5	-1	-1	+1	
6	+1	-1	+1	
7	-1	+1	+1	
8	+1	+1	+1	

## (b) EXPERIMENTAL PLAN FOR STUDY OF $p$

The fission gas internal pressure is affected by two input parameters only. So a full factorial experiment here requires only four runs (L-4) with the two inputs at two levels each. Again, this is done for nominal values, average values and standard deviation of the response  $p$ . The experimental design is shown in Table 5.2.2.

Table 5.2.2 L-4 Orthogonal array for 2 factor 2 level design

Run #	A	B	output
1	-1	-1	
2	+1	-1	
3	-1	+1	
4	+1	+1	

The data can again be analysed in two ways ( for location and variance respectively ) as given above in Section 5.2(a).

## 5.3 RESULTS AND THEIR INTERPRETATION

Tables 5.3.1 and 5.3.2 give the results of the orthogonal array experiments for nominal and average values and for standard deviation of the four output parameters of interest here. Figures 5.3.1 to 5.3.19 give the distribution of the output parameters for the different factor combinations considered under the orthogonal arrays. The last row in both the tables corresponds to the calculations for the existing design parameter settings of input parameters [11].

The results for gap conductance( $h_g$ ) show that maximum values

for nominal and average values are obtained for the first two rows of the array. Since  $h_g$  depends on two factors ( $d_p$  and  $d_i$ ) only, the latter four rows are a mere repetition of the first four. The third row of the array shows a very low value and this low gap conductivity is not very acceptable as this grossly impedes the flow of heat from fuel to coolant. Consequently, this increases the fuel center-line temperature.

About the values of the standard deviation, it is lowest for third row, which also shows lowest values of  $h_g$ . It is moderate for fourth row and high for the first two rows. Generally, the standard deviation values are much higher than average values because in many cases simulated pellet-cladding interference occurs; these cases are assigned a very high  $h_g$  value to facilitate complete calculations. This makes the overall distribution of  $h_g$  skewed rather than bell-shaped. Also, a comparison of the relative effect of input factors (Fig. 5.3.20) indicates that inner clad diameter has a greater effect on nominal values, average values, and standard deviation of the gap conductance than does fuel pellet diameter.

It is well known that fuel center-line temperature affects melting of fuel elements near the center-line, an undesirable operating condition. The orthogonal array experiments indicate that except for the third and seventh combinations, the other six sets of values are acceptable as nominal and average values. The standard deviation is maximum for third and seventh combinations of inputs. The comparison of factor effects show that outer clad diameter has a small effect on fuel center-line temperature (Fig. 5.3.21). The inner clad diameter has greatest effect on nominal and average values, and on the standard deviation of fuel center-line temperature. Also percentage of cases with center-line

temperature exceeding 2600 K is most strongly dependent on the inner clad diameter ( $d_i$ ) and is only weakly dependent on the outer clad diameter ( $d_o$ ) as shown in Fig.5.3.22.

The clad maximum temperature is seen to have a rather narrow distribution and its variation in nominal and average values is relatively small for different combinations of inputs. The comparison of factor effects (Fig.5.3.23) show that the inner clad diameter has a greater effect on clad maximum temperature than the outer clad diameter.

The fission gas internal pressure ( $p$ ) is seen to have large values for the first two combinations of average fuel grain radius and volume available for fission gases. The standard deviations are also large for these combinations. The reason is that for many Monte-Carlo combinations corresponding to the first two rows, the volume available becomes negligible; a large value is assigned to complete the calculation for internal pressure for these combinations. The comparison of factor effects (Fig. 5.3.24) shows that the volume available ( $V$ ) for fission gases has a markedly large effect on the nominal and average values of  $p$  (the fission gas internal pressure) and on its standard deviation than does the average fuel grain radius  $a$ . Also, the percentage of cases with fission gas pressure ( $p$ ) exceeding 95 bar has a strong dependence on volume available for fission gases and a lesser dependence on average fuel grain radius as shown in Fig.5.3.24.

#### 5.4 DISCUSSION

From the various figures showing the distribution of gap conductance ( $h_g$ ) it follows that the first two rows (experiments) give acceptable distributions of  $h_g$  while next two rows give relatively poor (unacceptably small) values of  $h_g$ . The first two

rows are also better than the existing design settings [9] of inputs (shown in the last row of Table 5.3.1). However, Row 2 represents interference of pellet and cladding even at nominal settings, which is undesirable. Thus the first row may provide a superior performance replacement for input parameter settings in use [11] as far as  $h_g$  is concerned.

The distribution diagrams for the center-line temperature ( $T_c$ ) show that first, second, fifth and sixth input factor combinations (rows) would give reasonably acceptable values of  $T_c$ , whereas rest of the combinations give undesirably high  $T_c$  values. These acceptable cases also yield  $T_c$  values and distributions that are superior to those at the existing nominal settings of input parameters[11]. Further, although second and sixth combinations of inputs yield the best results (  $T_c$  distributions with small average  $\sim 2000$  K and low standard deviation  $\sim 55$  K) they are accompanied by interference and the consequent stressing of pellet and cladding. Therefore, the first and fifth combinations of inputs would appear to provide the best results for fuel center-line temperature performance, better than the existing combinations of the inputs.

To reduce the interference of pellet and cladding, the tolerance limits especially for inner clad diameter could be tightened.

As far as clad temperature ( $T$ ) is concerned, it retains fairly safe values throughout and its values and their distributions do not alter much when inner and outer clad diameters change.

The distributions observed of the fission gas internal pressure ( $p$ ) convey the following. The first two rows (Table 5.3.2) would lead to many "high internal pressure" values, which

are undesirable. The next two rows give mostly acceptable results in terms of average values as well as standard deviation. These combinations are also better than those of existing combinations of input parameters [11].

An increase in the average fuel grain radius reduces the fission gas internal pressure. But volume (V) available for fission gases has a very strong influence on the internal pressure build-up and therefore V should be ensured to be enough so that the internal pressure does not attain values exceeding 95 bar.

dp	dl	dc	h-nom	h-av	sigma-h	Ic-nom	Ic-av	sigma-Ic	Ii-nom	Ii-av	sigma-Ii
14.4027	14.4083	15.2424	17799	57452	257272	2083.4	2137.9	134.6	603.28	603.29	0.699
14.4227	14.4083	15.2424	100000	98567	423530	2027.3	2047.0	56.4	603.28	603.27	0.694
14.4027	14.4529	15.2424	1994	2601	6579	2622.7	2616.4	198.3	601.35	601.37	0.703
14.4227	14.4529	15.2424	3311	10308	55158	2379.8	2386.6	189.5	601.35	601.34	0.695
14.4027	14.4083	15.2624	17799	56262	165962	2084.2	2138.7	136.3	604.08	604.09	0.703
14.4227	14.4083	15.2624	100000	97838	431439	2028.1	2046.3	55.2	604.08	604.10	0.704
14.4027	14.4529	15.2624	1994	2473	3613	2623.5	2622.1	197.9	602.15	602.15	0.694
14.4227	14.4529	15.2624	3311	9363	27053	2380.6	2382.0	191.0	602.15	602.16	0.700
14.4127	14.4306	15.2524	5582	33367	306225	2232.2	2248.0	174.2	602.72	602.72	0.695

TABLE 5.3.2 PERFORMANCE ANALYSIS RESULTS OF P

a	V	p-nom	p-avg	sigma-p
48.500	1.151	53.54	59.73	33.5
51.500	1.151	50.69	57.92	73.3
48.500	2.135	28.85	29.62	4.9
51.500	2.135	27.32	27.90	4.7
50.000	1.643	36.47	38.32	9.0

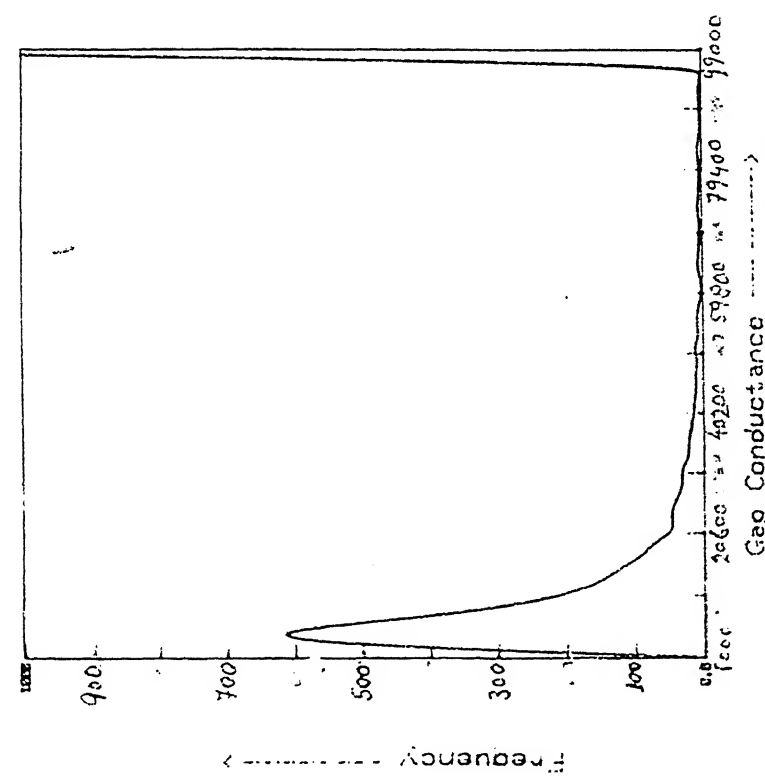


Fig. 5.3.1 h - Distribution For Run 1

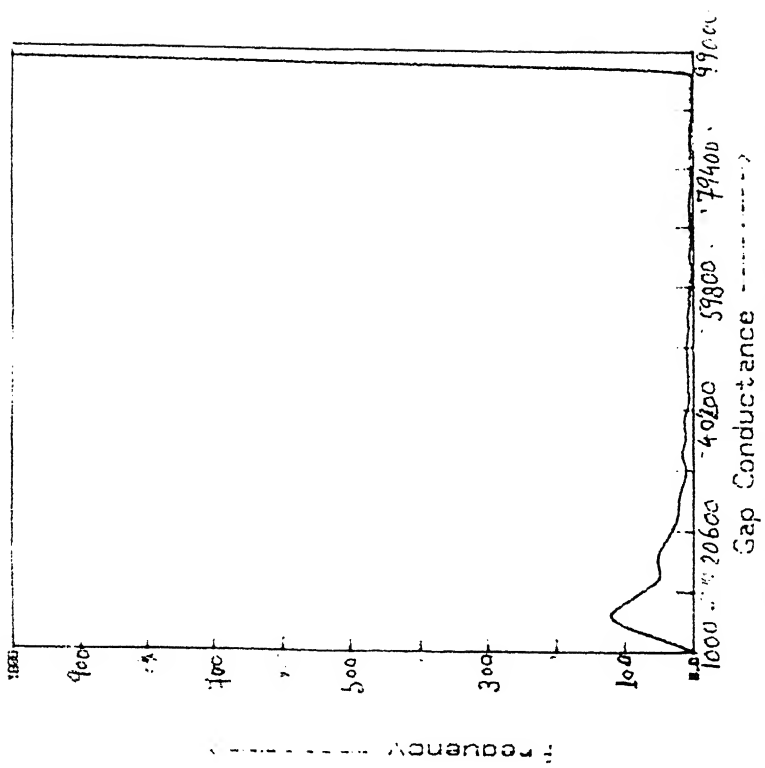
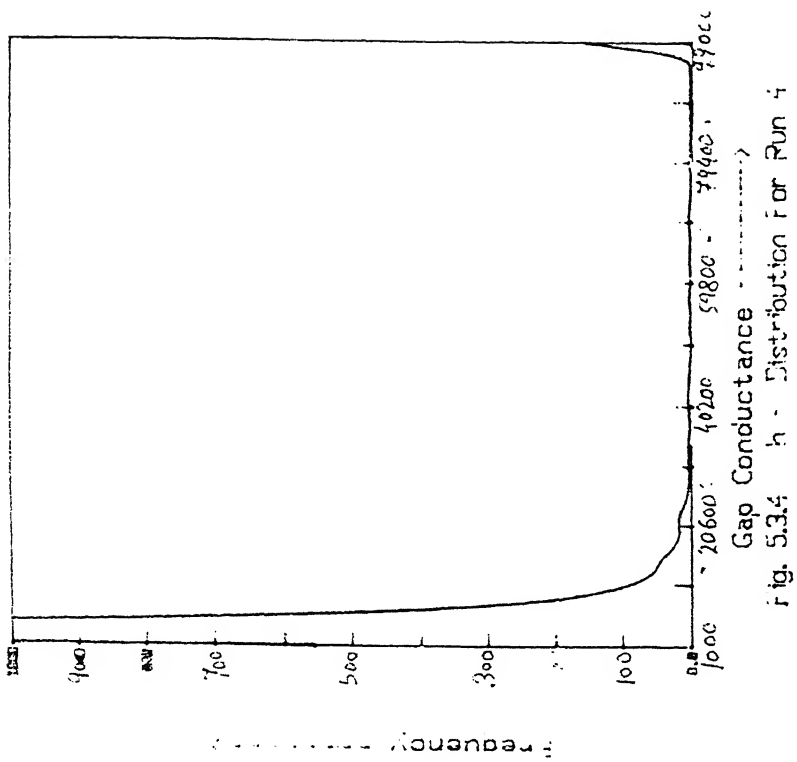
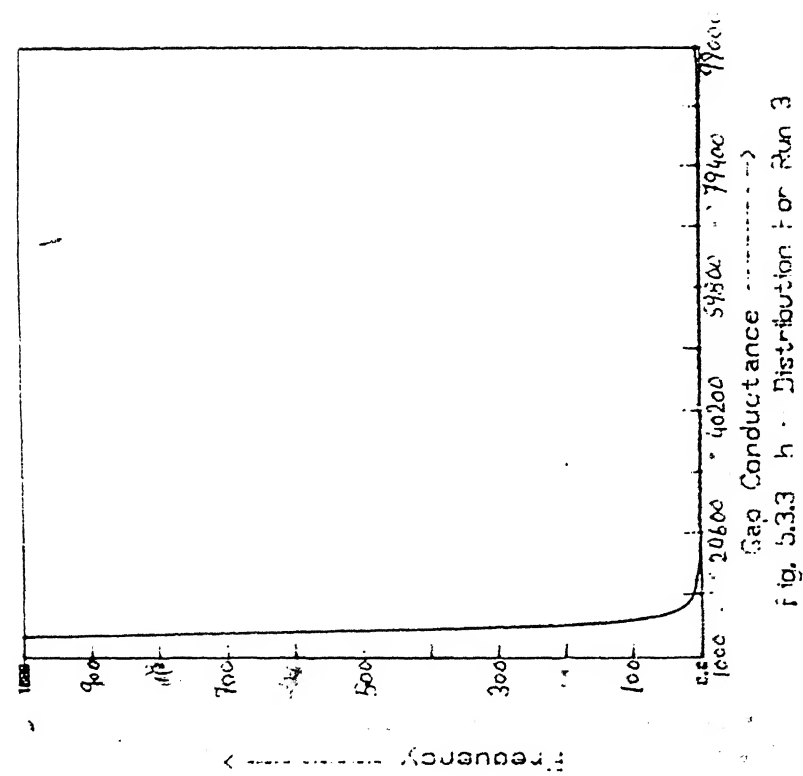
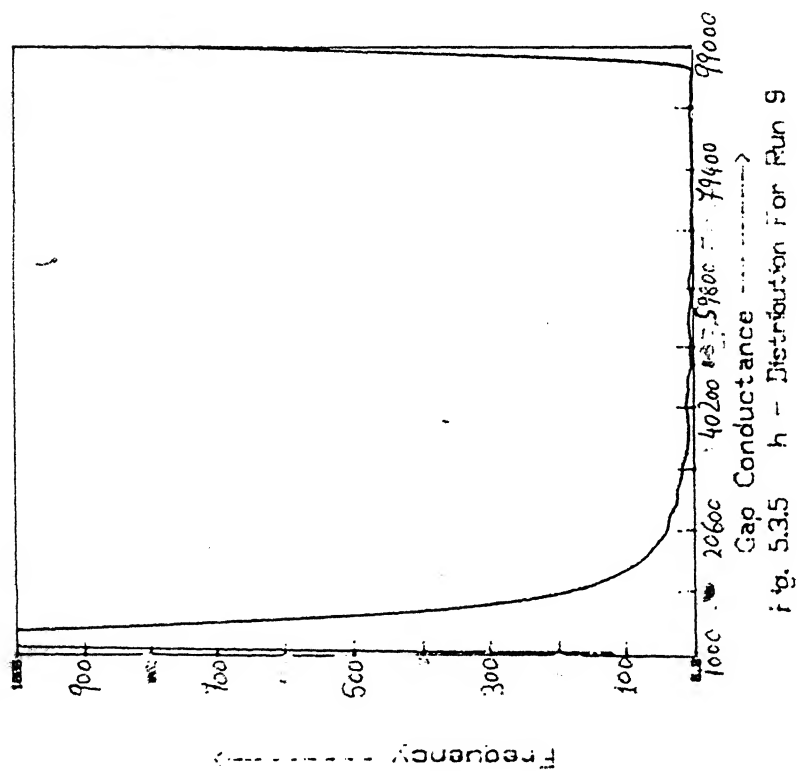
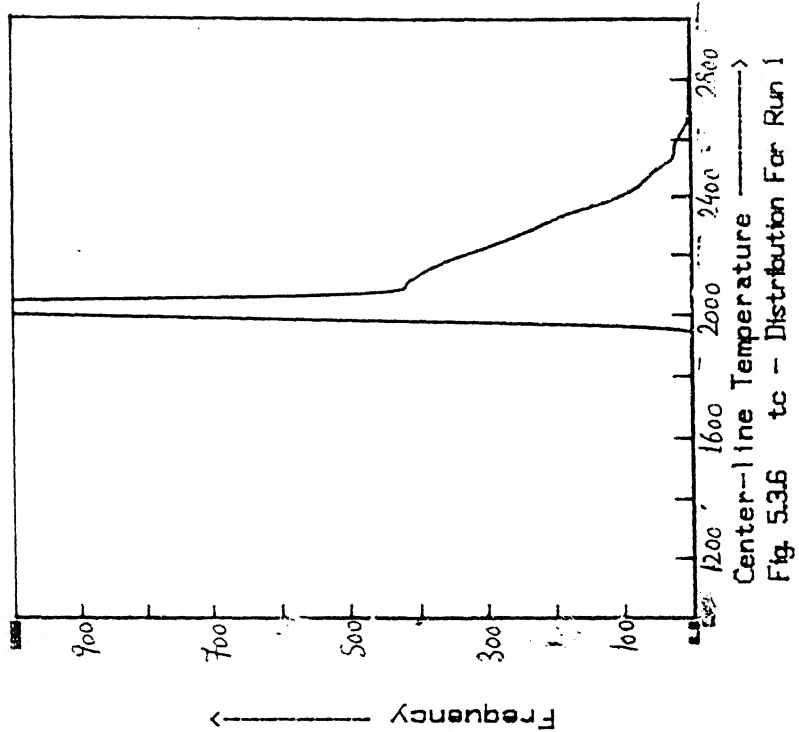
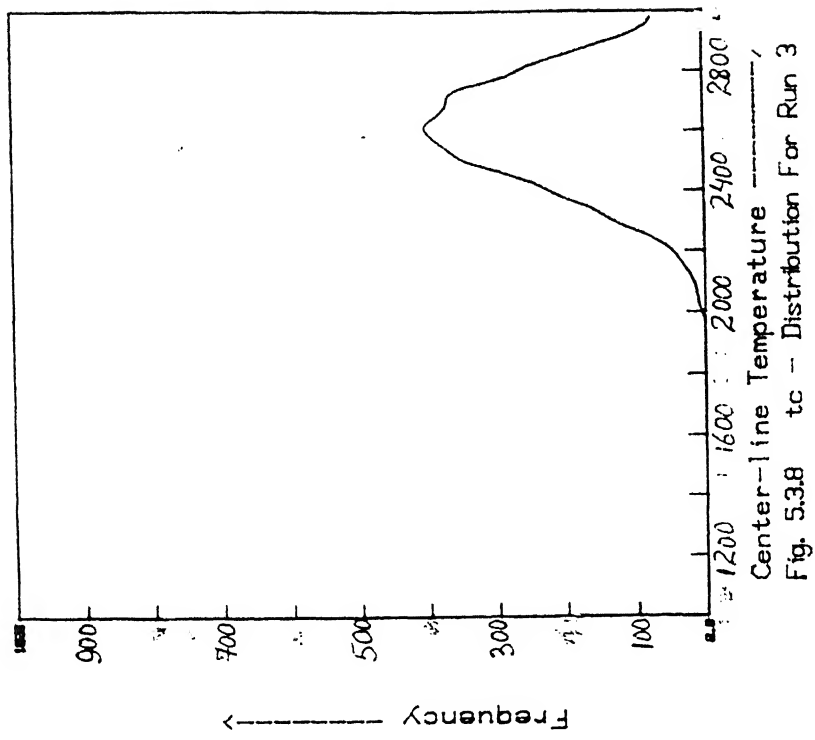
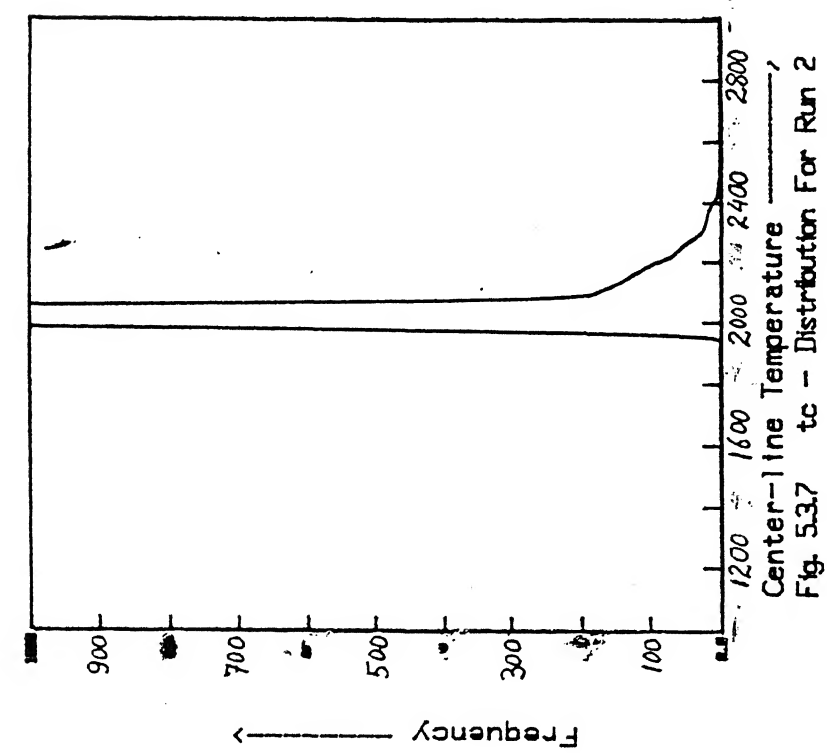
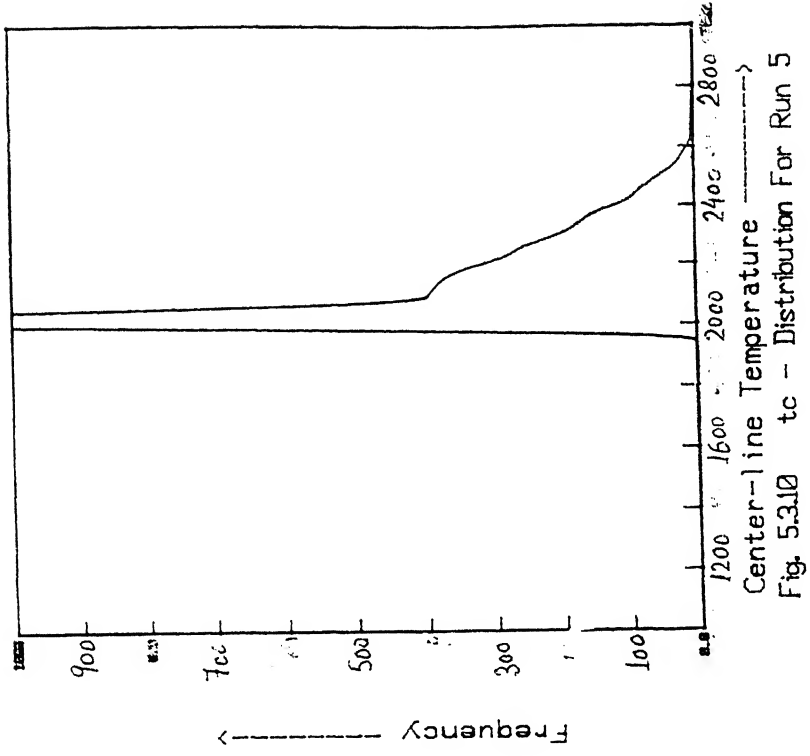
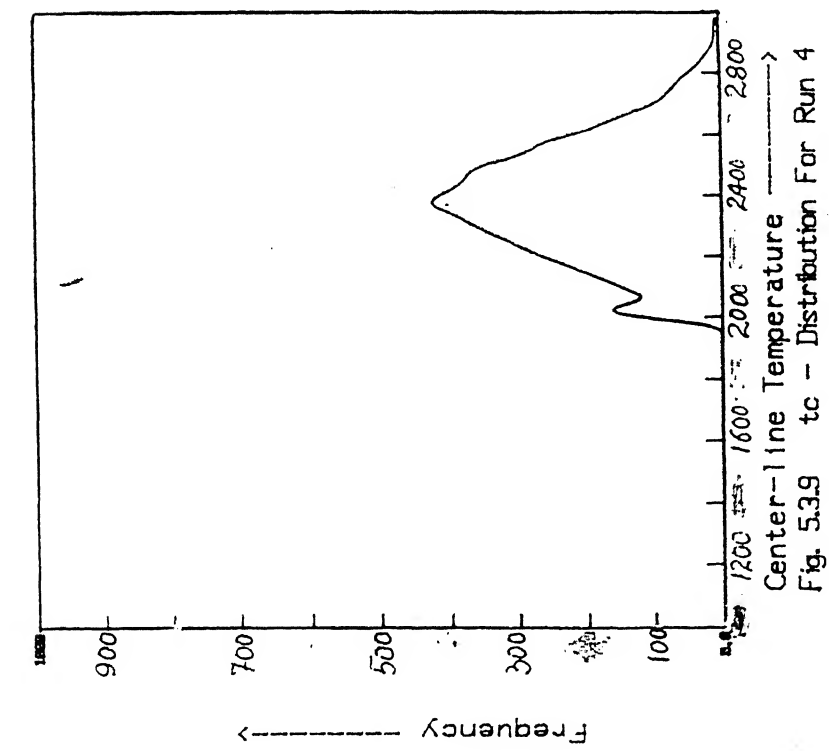


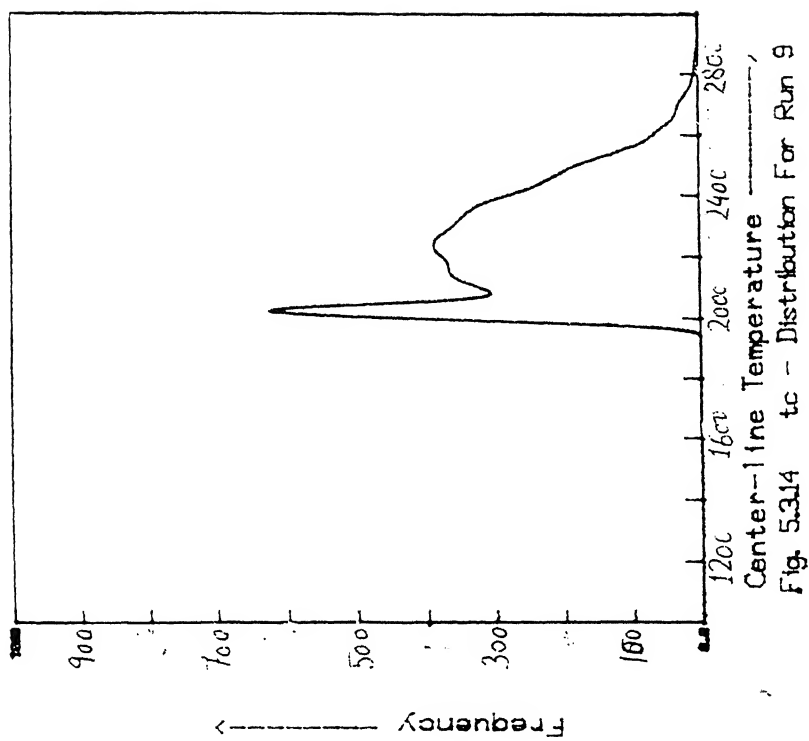
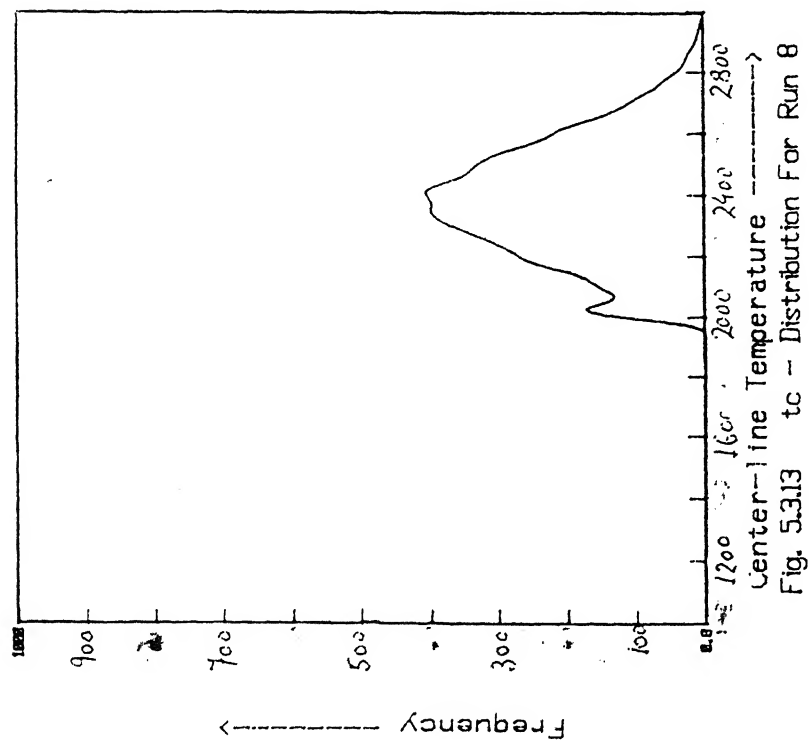
Fig. 5.3.2 h - Distribution For Run 2











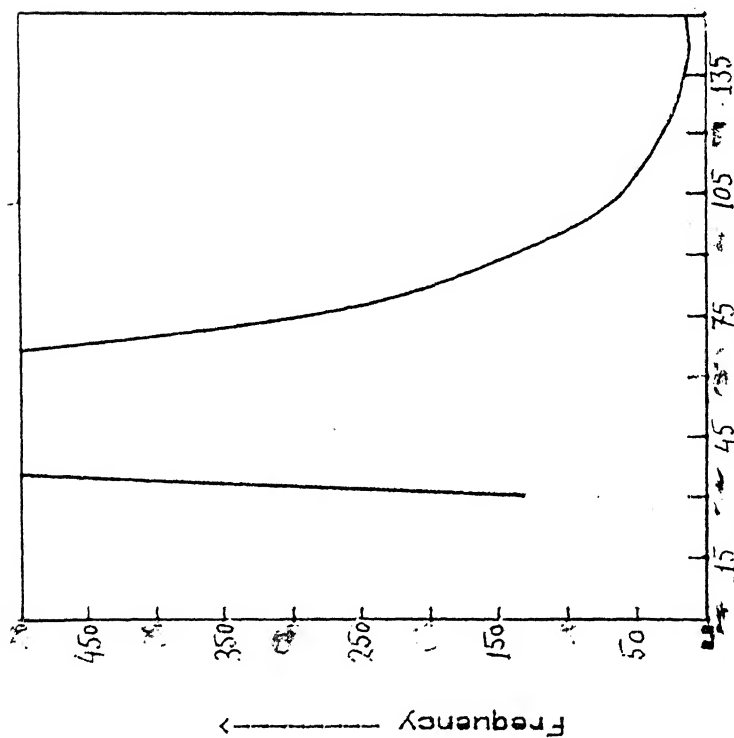


Fig. 5.3.15 p - Distribution For Run 1

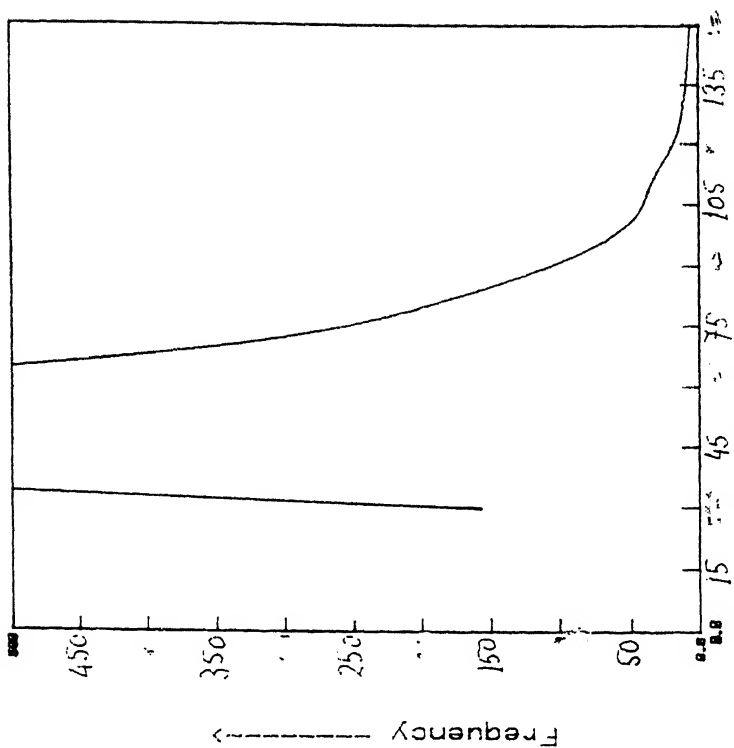
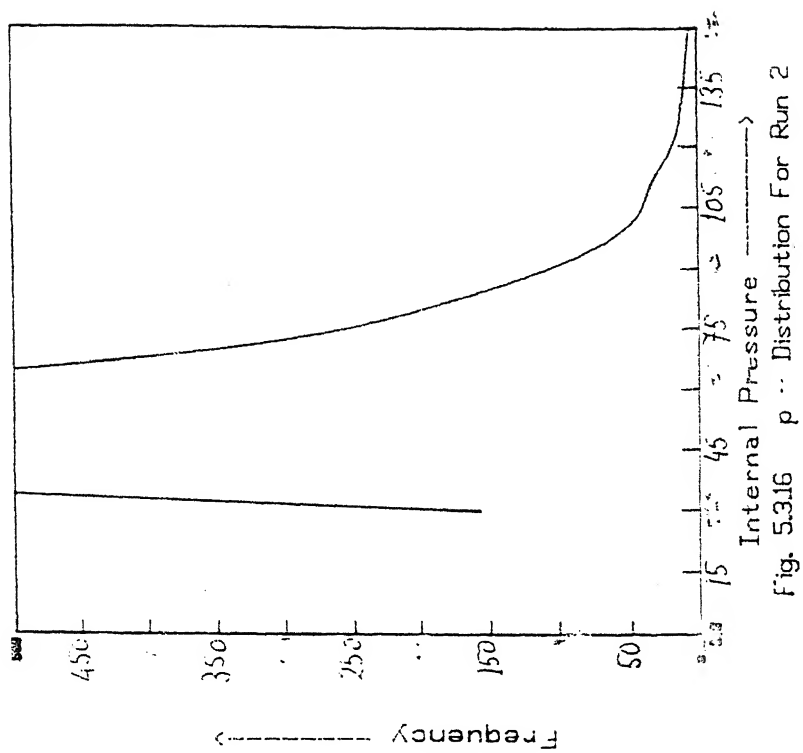
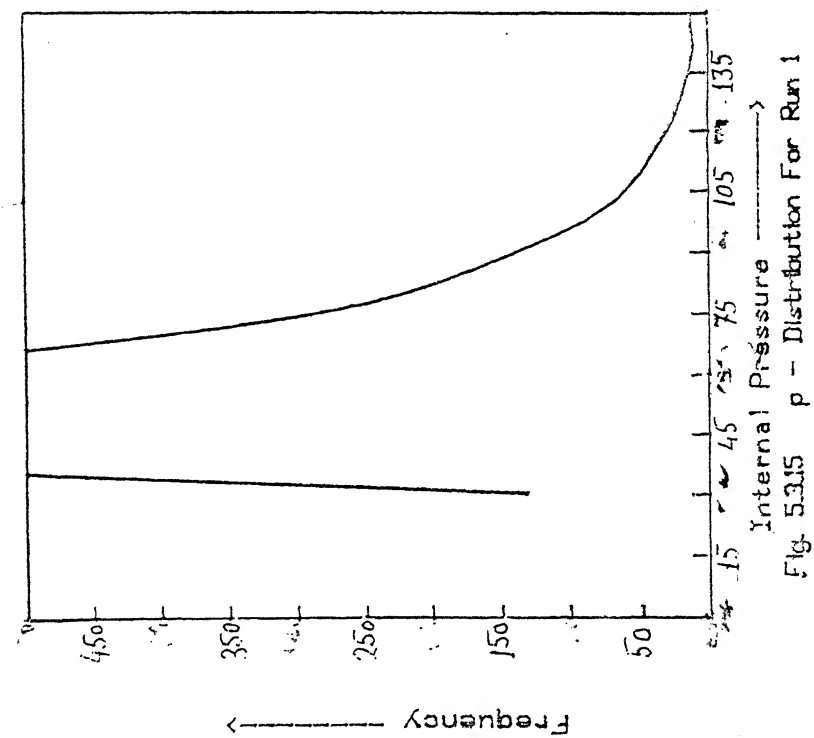


Fig. 5.3.16 p - Distribution For Run 2



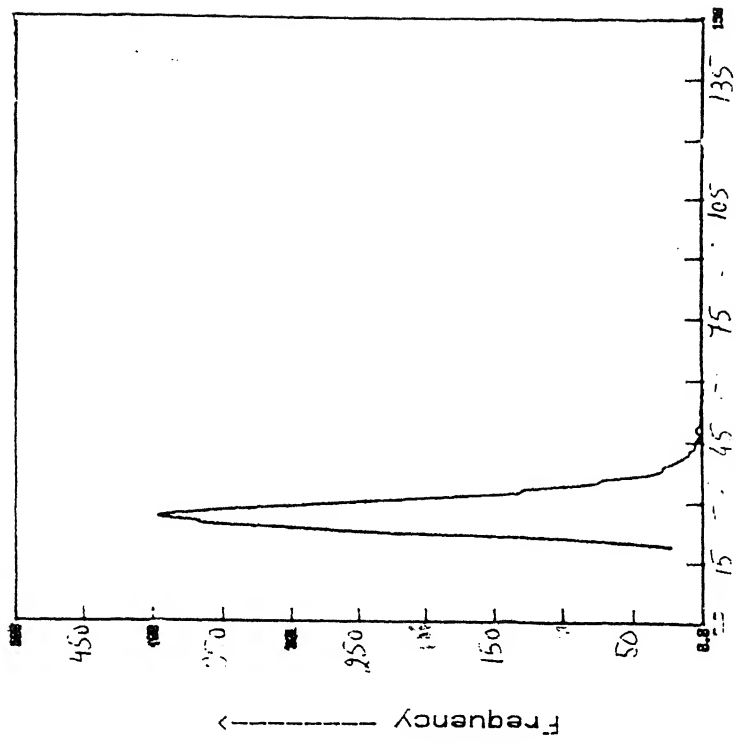


Fig. 5.3.18 p - Distribution For Run 4

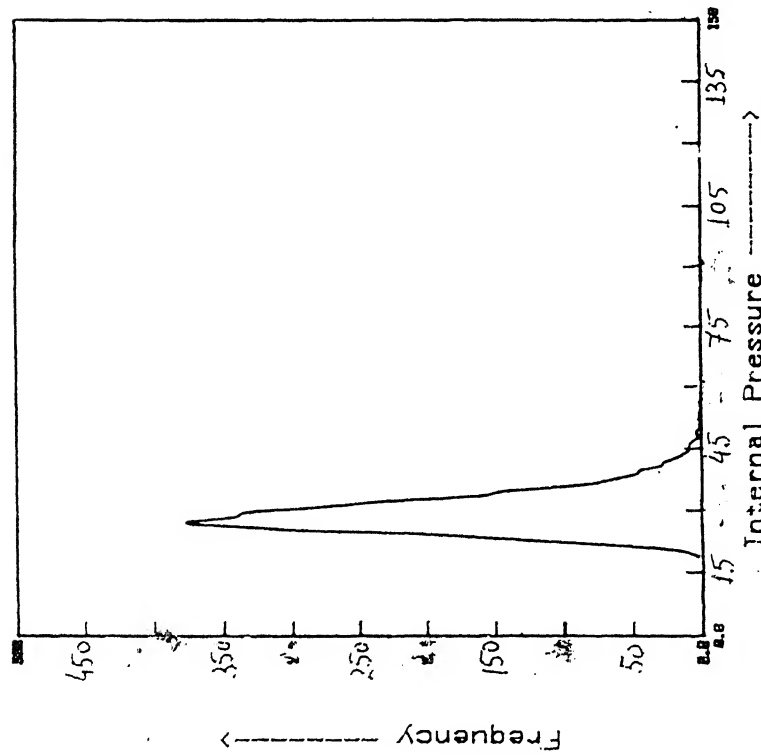


Fig. 5.3.17 p - Distribution For Run 3

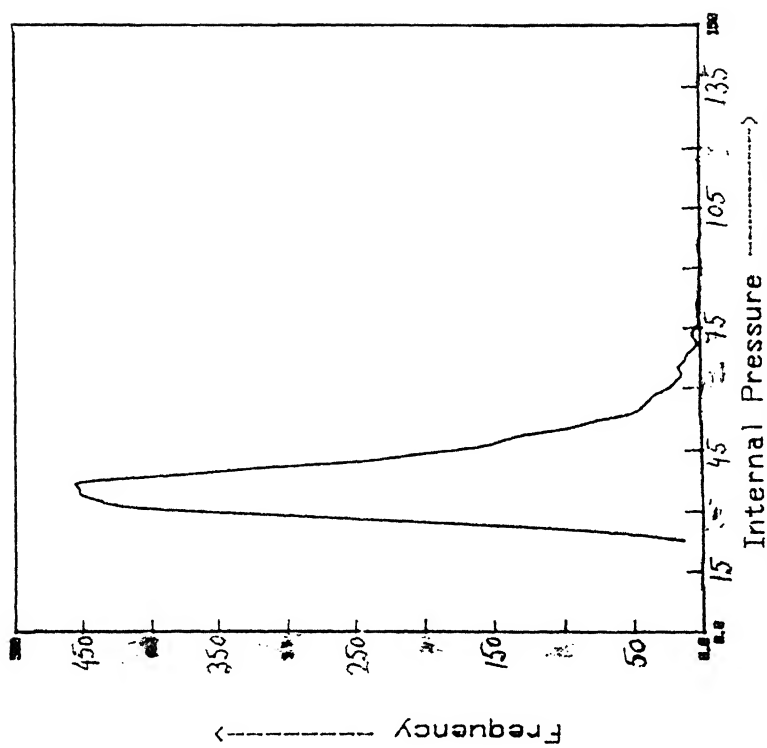


Fig. 5.3.19 p - Distribution For Run 5

Fig 5320 DEPENDENCY OF h ON dp, di and dc

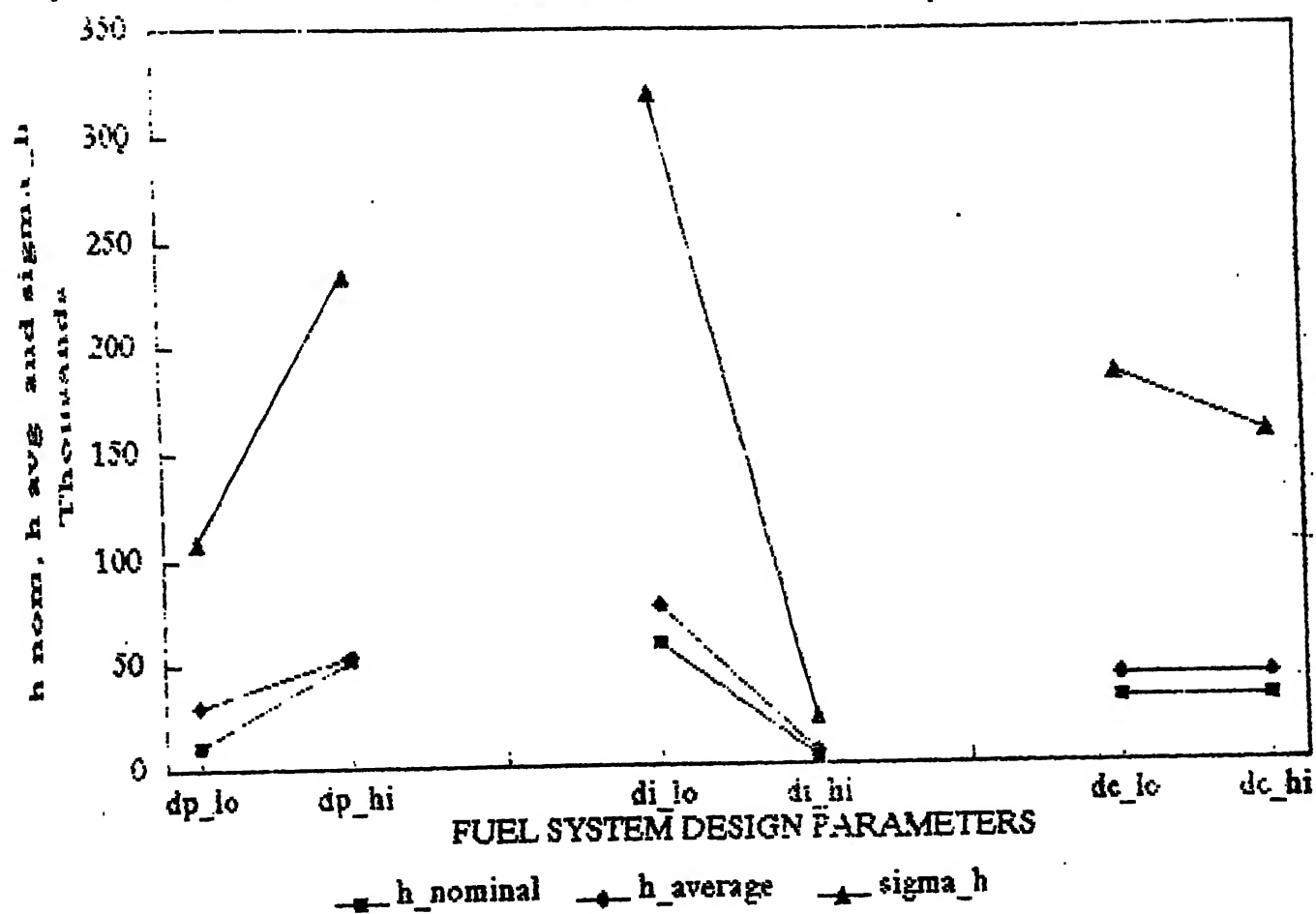


Fig. 5.321 DEPENDENCY OF Tc ON dp, di and dc

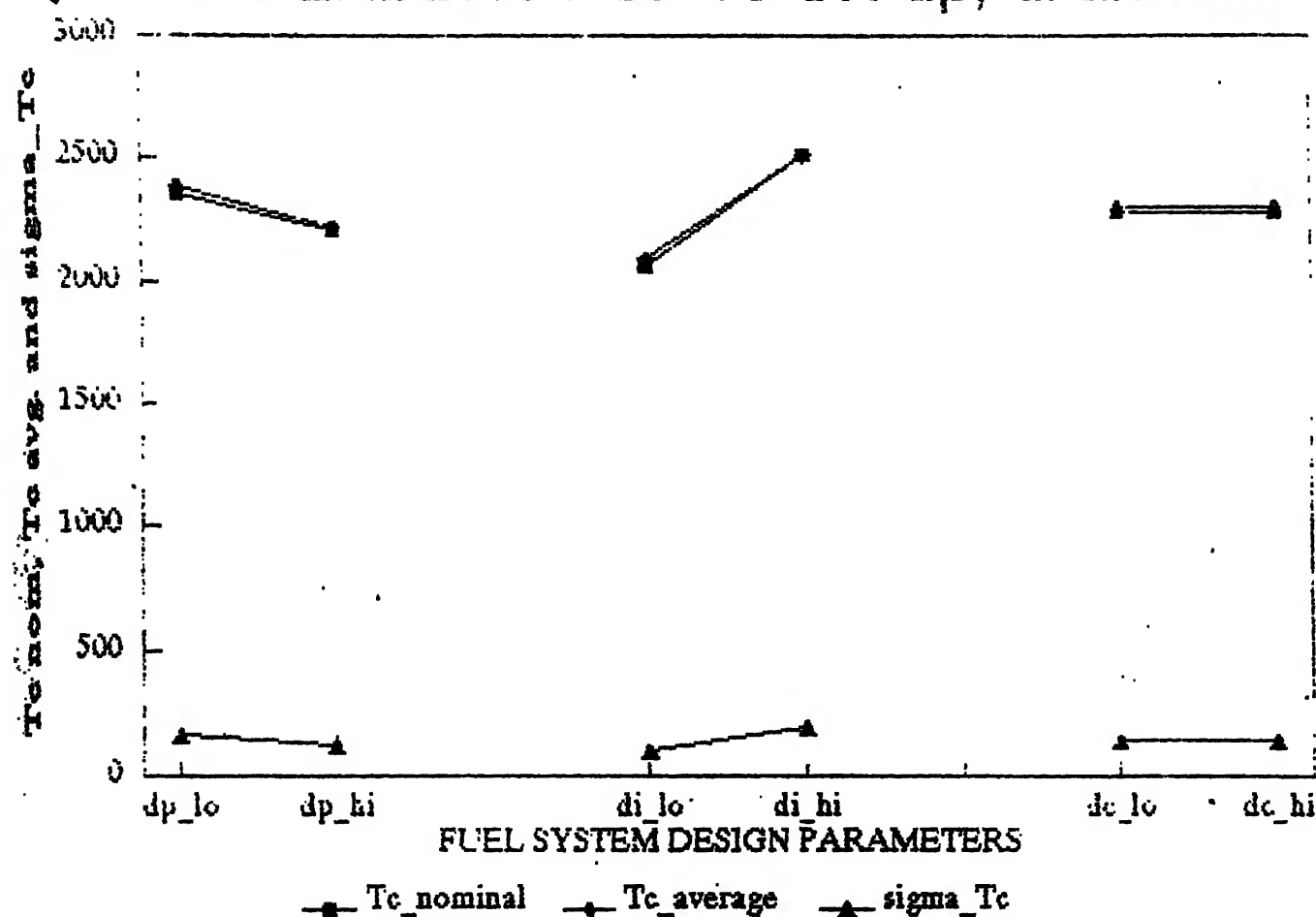


Fig 5-3-22 DEPENDENCY OF % PELLETS WITH  $T_c > 2600\text{ K}$  ON  $dp$ ,  $di$  and  $dc$

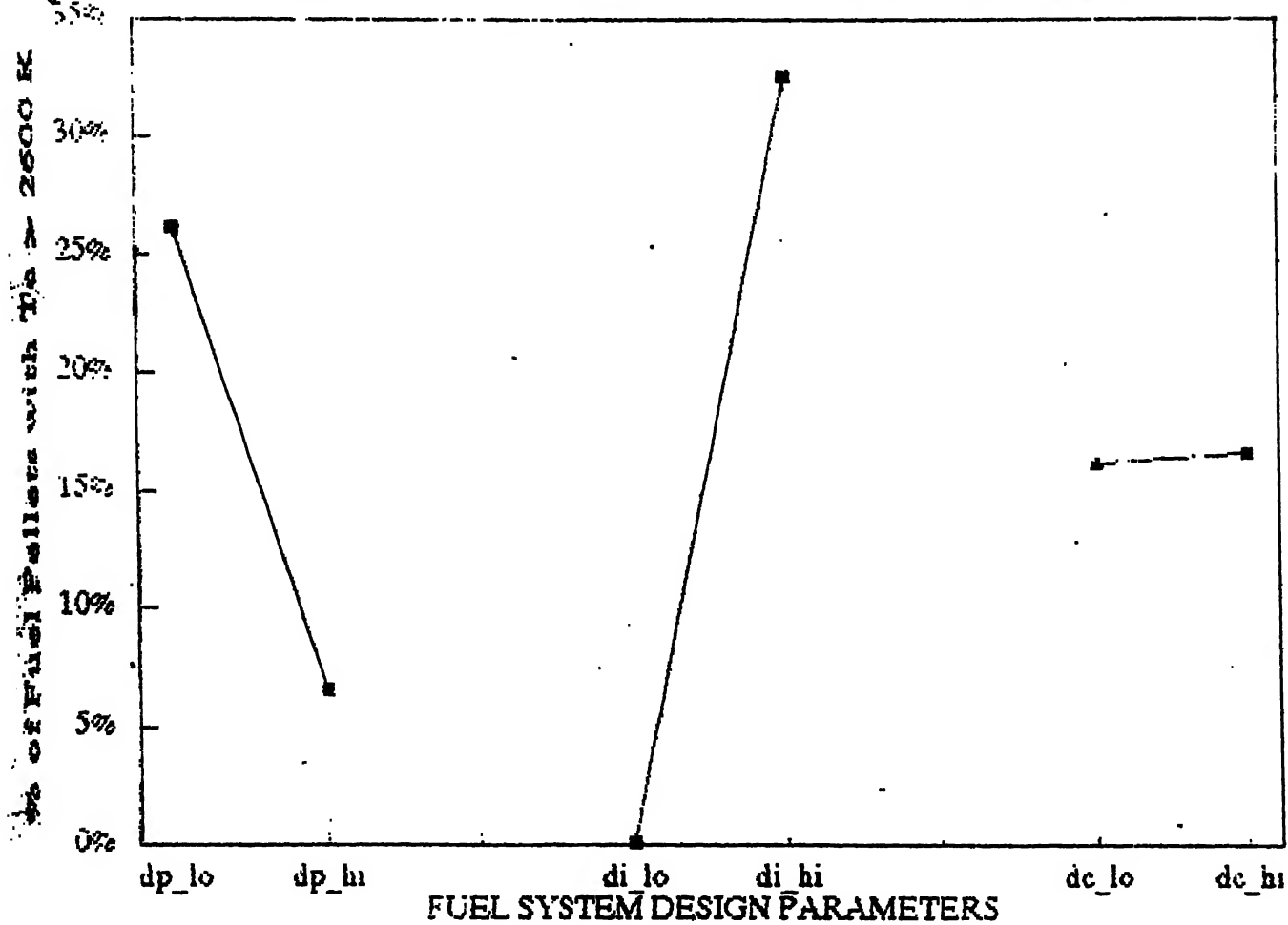


Fig 5.3.23 DEPENDENCY OF TI ON dp, di and dc

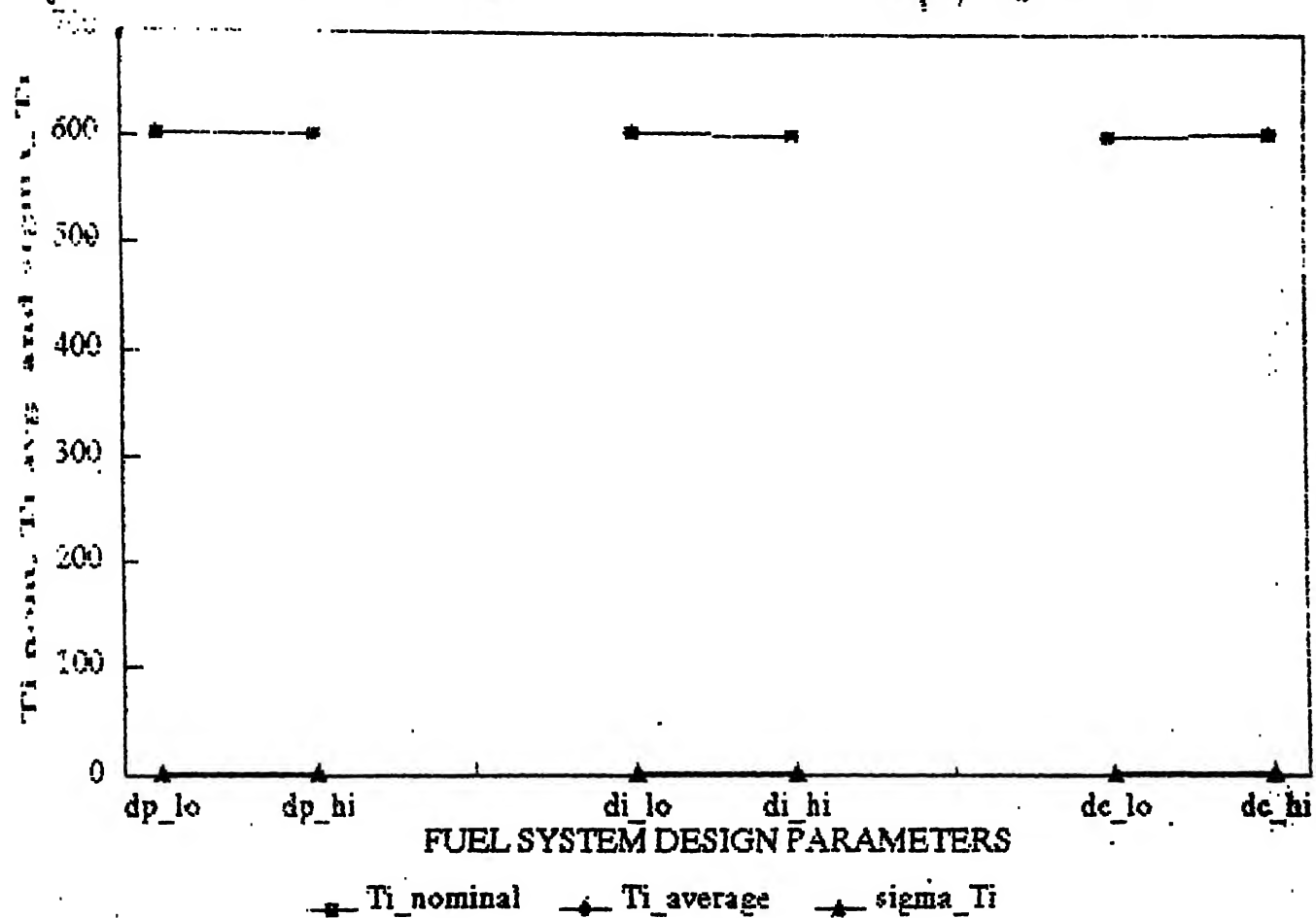
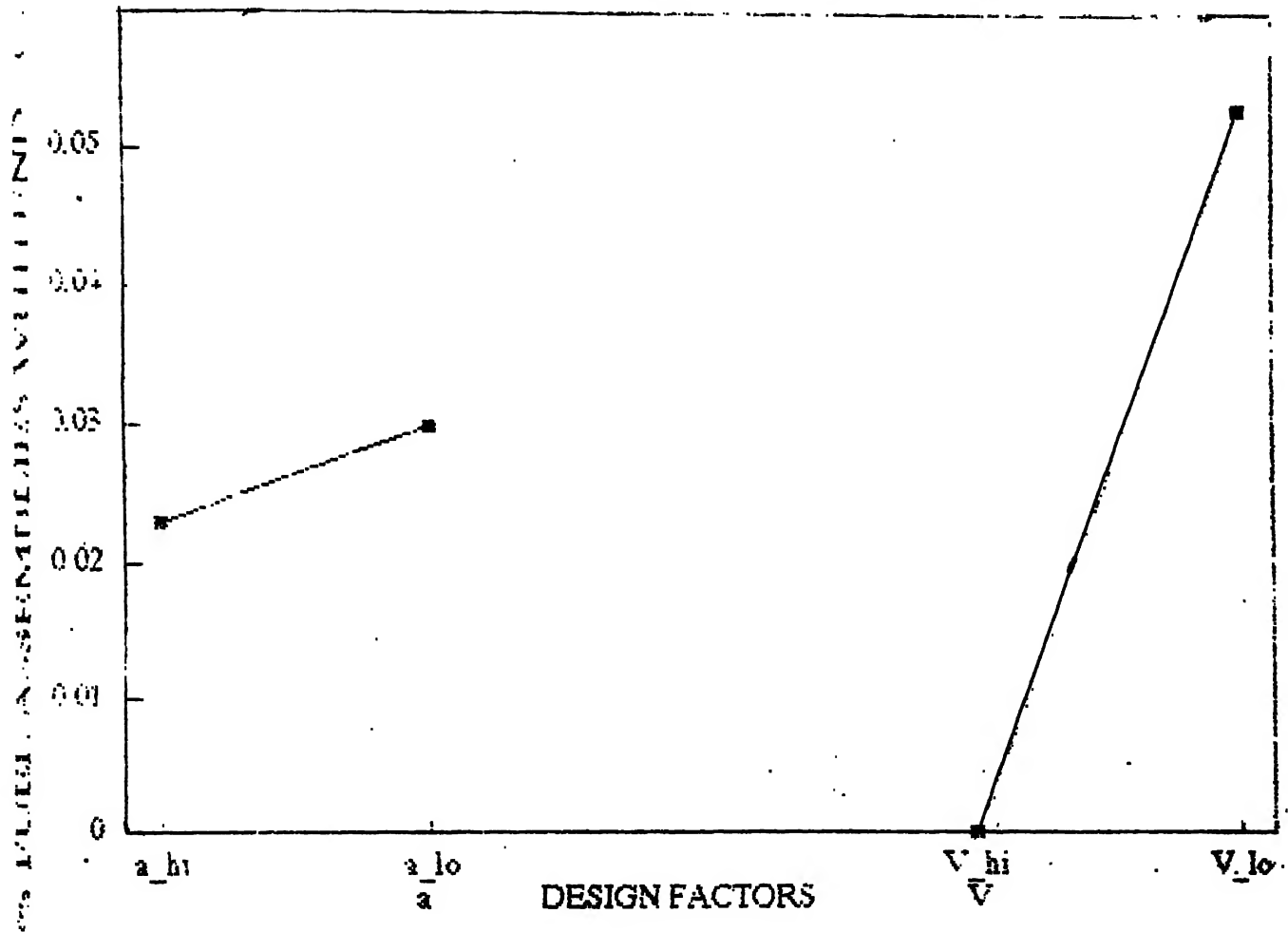


Fig. 5.3-24 DEPENDENCY OF FUEL ASSEMBLIES WITH  $p > 95$  ON  $a$  and  $V$



## CHAPTER 6

## CONCLUSIONS AND SUGGESTIONS FOR FUTURE WORK

## 6.1 CONCLUSIONS

Tables 6.1 and 6.2 show some of the conclusions which can be drawn from the probabilistic and deterministic analysis of the fuel elements as presented in the previous chapters of this work.

## SWELLING PLAYS AN IMPORTANT ROLE

The calculations indicate that the fuel swelling plays a very important role in determining gap conductance, fuel center-line temperature and fission gas internal pressure in a fuel rod. Swelling has a negative effect on pellet-clad interaction and stressing. It also increases the fission gas pressure in the fuel pin by reducing the volume available for fission gases escaping from the pellets. Nevertheless swelling is found to reduce the fuel center-line temperature and increase the gap conductance which are desirable effects. Fractional fission gas release directly affects fission gas internal pressure and indirectly but significantly affects gap conductance and fuel center-line temperature through gap conductivity. Also it is itself quite influenced by average fuel grain radius and is weakly dependent on pellet density.

## TIGHTER TOLERANCES IMPROVE GAP CONDUCTANCE AND CENTER-LINE TEMPERATURE

Among as-fabricated parameters inner clad diameter is the most crucial one. Especially with its large tolerance it plays a more important role in the variation of the gap than the pellet diameter and affects significantly the pellet-clad interference.

TABLE 5.1 DETERMINISTIC RESULTS

	2% swelling	3% swelling	4% swelling
Nominal values of fractional fission gas release( % )	11.36 %	11.36 %	11.36 %
Nominal values of gap conductance in W/sqr.m/K	1530.36	5581.66	16.00.000 highest value
Nominal value of inner clad surface temperature( K )	602.72	602.72	602.72
Nominal value of fuel centre-line temperature( K )	2810.93	2232.20	2015.33
Nominal value of fission gas pressure (bar )	27.51	36.53	67.73

TABLE 5.2 PROBABILISTIC RESULTS

% of area of inner clad in contact with fuel pellet	2.26%	29.16%	81.59%
% of fuel rods melting near the centre line	29.71%	2.26%	0.01%
% of fuel rods with fission gas pressure > coolant pressure	0.25%	6.01%	35.1%

fission gas pressure, gap conductance and fuel center-line temperature. Pellet diameter, density of fuel in the element and mass of fuel in each element are other important parameters. Due to large tolerances of mass and density of fuel, there is large variation in fission gas pressure and for certain combinations it becomes very large. So it is desirable to have low tolerance limits for inner clad diameter, mass and density of fuel. The former would reduce pellet clad interaction without affecting other performance parameters negatively. Reducing the variation of pellet density and fuel mass in each element would decrease the large variation in fission gas pressure especially the large pressure values, which occur if tolerance limits are high (i.e. as in present work). The tolerance values in clad length and outer clad diameter can be somewhat relaxed without affecting the parameters studied here to a significant level.

#### NOMINAL SETTINGS OF INPUT PARAMETERS

From the orthogonal array experiments, altering the nominal settings of the input parameters were found to give better distributions of gap conductance and fuel center-line temperature (average values as well as standard deviations) as discussed in Section 5.4. The melting of fuel elements near the center-line reduces for some combinations of inputs (pellet diameter, inner clad diameter and outer clad diameter).

#### VOLUME AVAILABLE GREATLY AFFECTS INTERNAL PRESSURE

Volume available for fission gases is found to have a stronger effect on fission gas pressure than average fuel grain radius and can be more effectively used to keep the internal pressure low.

#### STATISTICAL ANALYSIS PROVIDES IMPROVED INSIGHT INTO THE FUEL

## SYSTEM'S BEHAVIOR

The present work has been able to generate more insight into the fuel system's behavior under manufacturing/assembly uncertainties than what is achieved by worst case analysis. Examples are quantification of interference percentage, melting percentage and high internal pressure percentages and factor effects in the response (output) variables.

The analysis presented here is for particular bundles of fuel rods with linear thermal power rating of  $55 \text{ kWm}^{-1}$  and fuel burn-up of 12,500 MWD per ton of U-metal under constant operating conditions. These are one of the worst affected fuel elements in a PHWR, if transient effects are ignored, including any transients in the coolant flow rate. Similar analysis for other group of fuel rods (or bundles) and their proper superposition can lead to statistical analysis of complete core of a reactor.

## 5.2 SUGGESTIONS FOR FUTURE WORK

The present work is a simple but perhaps reasonably realistic analysis of fuel rod behaviour in a reactor core and there is a lot of scope for future work in the area of statistical performance analysis/evaluation of PHWR in-core fuel. A few suggestions are listed below:

(1) Actual input distribution functions from sample tests can be used for input distributions instead of assuming them normal, as has been done in the present work. Further, distributions/variations can be introduced in other input variables e.g., coolant-clad heat transfer coefficient which may vary due to constriction, obstructions etc.

(2) Iterations can be introduced in recalculating the inputs which are affected by certain outputs, e.g., fission gas release

can be made to depend on the pellet temperatures or gap conductivity, since both of them depend on the fission gas released. This would improve the model's predictability.

(3) Other modeling parameters can be introduced in calculations of outputs such as cladding creep, stressing and cracking of pellet, etc.

(4) Some additional factors also affect the models. For instance, gas bubbles (which are formed due to the fission gases trapped in the fuel) affect fission gas release, structural changes affect properties of fuel, etc. Some verification (physical) experiments could strengthen the results. Established fuel rod codes may also be used along with response surface and regression analysis. In the present analysis, we used one-dimensional fuel rod model, and made additional assumptions as mentioned earlier. This simplified model allowed direct Monte Carlo calculations which might otherwise become computationally too expensive.

(5) Transients, accident and over-power conditions may also be brought under analysis by introducing the related effects into the models used.

(6) As the models are improved along the lines suggested above, other output parameters affecting the behaviour of fuel rods like chances of cracking of fuel or stressing of cladding and its failure chances, etc., can be evaluated.

(7) Such analysis can be extended to various groups of fuel bundles in a reactor and then their superposition can be done to describe the statistical performance of the reactor core as a whole.

(8) Factor effects on the response parameters and their variability estimated using realistic models as indicated above and statistically designed experiments can lead to improved

selection of fuel system design parameters.

With improvements in the quality of the models and the input fabrication and power history data, the probabilities of undesirable events can be predicted reliably and this would assist in the improved understanding of the actual behaviour of fuel rods in a reactor. However, many of these aspects (for example, model development) are areas of current research. Accurate models to describe various phenomena in the fuel rod both under steady state and transient conditions are yet to be established.

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```

program example (input,output,f1,f2,f3,f4,f5,f6,f7);
(this program calculates gap conductance & CL temp & inner clad temp)
(for given inputs which are normally distributed )
const
    pi=3.1415926;
type
    arr1=array[1..75] of real;
    arr2=array[1..80000] of real;
    arr3=array[1..60] of integer;
var
    f1,f2,f3,f4,f5,f6,f7:text;
    h,i,h1,h2,n1,n2:integer;
    g1,g2:arr1;
    dp,di,dc,hg,Tc,Ti:arr2;
    m1,s1,m2,s2,m3,s3,m4,s4,m5,s5,m6,s6,m7:real;
    e1,e2,e3,e4,e5,e6,e7,e8,e9:arr3;
    dpg,dig,dcg,hgg,Tcg,Tig:arr1;
    kg,kp,kc,Tcool,hc,fs,q1,NH,NT:real;
    maxh,maxT,minh,minT,mxTi,mnTi:real;
    mxdp,mndp,mxdi,mndi,mxdc,mndc:real;
    hav,Tcav,Tiav:real;

(-----)
function drand48:longreal; external c;
(This function generates random numbers uniformly distributed )
(between 0 & 1 through a function defined in the compiler )
(-----)

(-----)
procedure generate(m,s:real;i1,n:integer;a,b:arr1;var x:arr2);
(this procedure generates numbers which are normally distributed)
var
    j,k:integer;
    c,x1:array[1..80000] of real;
begin
    j:=0; k:=0;
    for j:=1 to n do begin
        c[j]:=drand48;
        if (c[j]<=0.00023) or (c[j]= 0.99977) then
            c[j]:=drand48;
        end;
        for j:=1 to n do begin
            for k:=1 to i1 do
                if (c[j]>b[k]) and (c[j]<b[k+1]) then begin
                    x1[j]:=a[k]+(c[j]-b[k])*(a[k+1]-a[k])/(b[k+1]-b[k]);
                    x[j]:=s*x1[j]+m;
                end;
            end;
        end;
    end;
end;
(-----)

(-----)
procedure distribution(n3,n:integer;x:arr2;var e:arr3;var max,min:real);
(this procedure obtains the distribution of a given set of )
(numbers into required no. of divisions )
var
    d:array[1..61] of real;
    l1,l2,l3:integer;
begin
    l1:=0; l2:=0; l3:=0; max:=x[l1]; min:=x[l1];
    for l3:= 1 to n do begin
        if ( x[l3] > max ) then max:=x[l3];
        if (x[l3] < min) then min:=x[l3];
        end;
        if max > (25*min) then max:=25*min ;
    end;
end;

```

```

for l2:=1 to n3 do begin
  d[l2]:=min+(l2-1)*(max-min)/n3 ;   e[l2]:=0; end;
d[n3+1]:=max;

for l1:=1 to n do
  for l2:=1 to n3 do begin
    if (x[l1] > d[l2]) and (x[l1]<d[l2+1]) then
      e[l2]:=e[l2]+1;
    end;
  end;
}
{-----}

{-----}
procedure graphdata(mx,mn:real ; n4:integer; var y:arr1);
(This procedure gives X-axis of a variable given max&min&#of divisions)
var
  i:integer;
  e:real;
begin
  e:=(mx-mn)/n4;
  y[1]:=mn+e/2;
  for i:= 2 to n4 do
    y[i]:=y[i-1]+e;
  end;
}
{-----}

begin (main program begins here )
  writeln(' Enter the no. of values required ');
  readln(n1);
  writeln('Enter the no. of divisions for distribution < 61');
  readln(n2);
  writeln(' Enter the swelling in % ');
  readln(fs);
  fs:=fs/100;
  m1:=0.01427; s1:=0.00001333;
  m2:=0.0144 ; s2:=0.00002974;
  m3:=0.01522; s3:=0.00001333;
  m4:=55000;
  m1:=m1*(1+fs/3);
  m2:=m2*(1+0.000006077*350);
  m3:=m3*(1+0.000006077*350);
  (This calculates mean values & std. deviations for inputs)

  reset(f1,'fxdata');
  rewrite(f2,'dish10');
  rewrite(f3,'dist10');
  (rewrite(f4,'dishdp1');)
  (rewrite(f5,'dishdi1');)
  (rewrite(f6,'dishdcl');)
  (rewrite(f7,'disti12');)

  i:=0; h:=0; h1:=0; h2:=0;
  while not (eof(f1)) do begin
    i:=i+1;
    readln(f1,g1[i],g2[i]);
  end;
  (This reads normal distribution from the input file fxdata)

  generate(m1,s1,i,n1,g1,g2,dp);
  generate(m2,s2,i,n1,g1,g2,di);
  generate(m3,s3,i,n1,g1,g2,dc);
  (This assigns normal distribution to the input parameters)

  kp:=3.1 ; kg:=0.05 ; kc:=14 ;
  (thermal conductivities of pellet/gap/clad ref.J.Belle&kaufmann)
  hc:=63460 ; (heat transfer coefficient of clad & coolant ref.AERB data)

```

```

Tcool:=550 ;(average coolant temp = 550 K ref. AERB data)
q1:=55000 ;(average linear power rating =55KW/m ref. AERB data)
if ( m2 > m1 ) then begin
m5:=(2*kg)/(m1*ln(m2/m1));
m6:=Tcool+m4/pi*(1/(4*kp)+(ln(m2/m1))/(2*kg)+(ln(m3/m2))/(2*kc)+1/(m3*hc));
end
else begin
m5:=1600000;
m6:=Tcool+m4/pi*(1/(4*kp)+1/(m1*m5)+(ln(m3/m2))/(2*kc)+1/(m3*hc)) ;
end;
m7:=Tcool+m4/pi*((ln(m3/m2))/(2*kc)+1/(m3*hc));
(This calculates nominal values for the two outputs)

NH:=0; hav:=0; Tcav:=0 Tiav:=0; NT:=0;
for h:=1 to n1 do
begin
hg[h]:=(2*kg)/(dp[h]*ln(di[h]/dp[h]));
if ( hg[h] < 0 ) then begin
hg[h]:=1600000;(to account for interference)
NH:=NH+1;
end;
(dp & di are given normally distributed variables & hg is )
(obtained as given by the model for the two with given coeffs.)
hav:=hav+hg[h] ;
Tc[h]:=Tcool+q1/pi*(1/(4*kp)+1/(dp[h]*hg[h])+ln(dc[h]/di[h])/(2*kc)+1/(dc[h]*hc));
Tcav:=Tcav+Tc[h];
if ( Tc[h] > 3023 ) then NT:=NT+1;(3023 K=Melting point of UO2)
Ti[h]:=Tcool+q1/pi*(ln(dc[h]/di[h])/(2*kc)+1/(dc[h]*hc));
Tiav:=Tiav+Ti[h];
end;
(This calculates different values of outputs for each set of inputs)

NH:=NH*100/n1;
NT:=NT*100/n1;
hav:=hav/n1;
Tcav:=Tcav/n1;
Tiav:=Tiav/n1;

distribution(n2,n1,dp,e1,mxdp,mndp);
distribution(n2,n1,di,e2,mxdi,mndi);
distribution(n2,n1,dc,e4,mxdc,mndc);
distribution(n2,n1,hg,e5,maxh,minh);
distribution(n2,n1,Tc,e6,maxT,minT);
distribution(n2,n1,Ti,e7,mxTi,mnTi);
(This obtains the distribution of all input & output variables)

graphdata(mxdp,mndp,n2,dpg);
graphdata(mxdi,mndi,n2,dig);
graphdata(mxdc,mndc,n2,dcg);
graphdata(maxh,minh,n2,hgg);
graphdata(maxT,minT,n2,Tcg);
graphdata(mxTi,mnTi,n2,Tig);
(This calculates X-axis of the input & output distribution graphs)

(for h1:=1 to n1 do begin )
(writeln(f2,hg[h1]:8:4); )
(writeln(f3,Tc[h1]:4:4); )
(end; )

(writeln(f2); )
(writeln(f3); )
(writeln(f2,'Below is the distribution of the gap conductances '); )
(writeln(f2); )
(writeln(f3,'Below is the distribution of the centre-line temp '); )
(writeln(f3); )

```

```

(writeln(f2,'gap conductance  '); )
(writeln(f3,'centre-line temp. '); )
(writeln(f2)); )
(writeln(f3)); )

for h2:=1 to n2 do begin
  writeln(f2,hgg[h2]:10:2,' ',e5[h2]:5);
  writeln(f3,Tcg[h2]:7:2,' ',e6[h2]:5);
  (writeln(f4,dpg[h2]:9:7,' ',e1[h2]:5);)
  (writeln(f5,dig[h2]:9:7,' ',e2[h2]:5);)
  (writeln(f6,dcg[h2]:9:7,' ',e4[h2]:5);)
  (writeln(f7,Tig[h2]:6:2,' ',e7[h2]:5);)
end;

writeln(f2,'Max gap conductance value is ',maxh:10:2,'W/sqr.m/K');
writeln(f2,'Min gap conductance value is ',minh:10:2,'W/sqr.m/K');
writeln(f2,'Percentage of area where contact b/w clad-pellet:',NH:5:2,' %');
writeln(f2,'Nominal value of gap conductance = ',m5:10:2,'W/sqr.m/K');
writeln(f2,'Average value of gap conductance = ',hav:10:2,'W/sqr.m/K');

writeln(f3,'Max centre-line temp is ',maxT:7:2,' K');
writeln(f3,'Min centre-line temp is ',minT:7:2,' K');
writeln(f3,'Nominal value of centre-line temperature = ',m6:7:2,' K');
writeln(f3,'Average value of centre-line temperature = ',Tcav:7:2,' K');
writeln(f3,'Percentage of fuel rods melting near centre ',NT:5:2,' %');

(writeln(f7,'Max inner clad temperature is ',mxTi:6:2,' K');)
(writeln(f7,'Min inner clad temperature is ',mnTi:6:2,' K');)
(writeln(f7,'Nominal value of inner clad temp = ',m7:6:2,' K');)
(writeln(f7,'Average value of inner clad temp = ',Tiav:6:2,' K');)

end.

```

```

program example (input,output,f1,f2,f3,f4,f5,f6,f7);
(this program calculates fuel rod internal pressure by the model)
(for given inputs(dp,di,L,a,m,d) which are normally distributed )
const
  pi=3.1415926;
type
  arr1=array[1..75] of real;
  arr2=array[1..80000] of real;
  arr3=array[1..60] of integer;
  arr4=array[1..60] of real;
var
  f1,f2,f3,f4,f5,f6,f7:text;
  h,i,h1,h2,n1,n2:integer;
  g1,g2:arr1;
  dp,di,L,a,p,m,d,f,V:arr2;
  m1,s1,m2,s2,m3,s3,m4,s4,m5,s5,m6,s6,m7,s7,m8,m9:real;
  e1,e2,e3,e4,e5,e6,e7,e8,e9:arr3;
  dpg,dig,Lg,ag,fg,Vg,pg:arr4;
  c,fr1,fr2,fr3,fr,fs,L1,R,Tg,V1,Pav,Vav,fav,NP,NV:real;
  maxp,minp,mxdp,mndp,mxdi,mndi,mxL,mnL,mxa,mna:real;
  mxf,mnf,mxV,mnV:real;

  (-----)
  function drand48:longreal ; external c ;

  (This functions generates random numbers uniformly distributed)
  (between 0 & 1 through a function defined in the compiler )
  (-----)

  (-----)
  procedure generate(m,s:real;i1,n:integer;a,b:arr1;var x:arr2);
  (this procedure generates numbers which are normally distributed)
  var
    j,k:integer;
    c,x1:array[1..80000] of real;
  begin
    for j:=1 to n do begin
      c[j]:=drand48;
      if ( c[j] <= 0.00023) or (c[j]>= 0.99977 ) then
        c[j]:=drand48;
      end;
      for j:=1 to n do
        begin
          for k:=1 to i1 do
            if (c[j]>b[k]) and (c[j]<b[k+1]) then
              begin
                x1[j]:=a[k]+(c[j]-b[k])*(a[k+1]-a[k])/(b[k+1]-b[k]);
                x[j]:=s*x1[j]+m;
              end;
            end;
          end;
        end;
      end;
    end;
  (-----)

  (-----)
  procedure distribution(n3,n:integer;x:arr2;var e:arr3;var max,min:real );
  (this procedure obtains the distribution of a given set of )
  (numbers in the given no.of equal divisions )
  var
    d:array[1..61] of real;
    l1,l2,l3:integer;
  begin
    l1:=0; l2:=0; l3:=0; max:=x[l1]; min:=x[l1];
    for l3:=1 to n do begin
      if ( x[l3] > max ) then max:= x[l3];
      if ( x[l3] < min ) then min:= x[l3];
    end;
  end;

```

```

        end;
        if (max > 25*min) then max:=25*min ;
        for l2:=1 to n3 do begin
            d[l2]:=min+(l2-1)*(max-min)/n3 ;
            e[l2]:=0 ;
        end;
        d[n3+1]:=max ;

        for l1:=1 to n do
            for l2:=1 to n3 do begin
                if (x[l1] > d[l2]) and (x[l1]<d[l2+1]) then
                    e[l2]:=e[l2]+1;
            end;
        end;
    }-----}

    {-----}
procedure fissiongas_release(a1,c1,T1:real; var f:real );
var
    logD,D:real;
begin
    logD:=-13.5-(10000/T1-5)*3/5;
    D:=0.0001*exp(2.303*logD);
    if (pi*pi*D*c1/(a1*a1) < 1) then
        f:=4*sqrt(D*c1/(pi*a1*a1))-1.5*D*c1/(a1*a1)
    else
        f:=1-a1*a1/(D*15*c1)+6*a1*a1/(sqr(sqr(pi))*D*c1)*exp(-sqr(pi)*D*c1/sqr(a1));
    end;
}-----}

    {-----}
procedure graphdata(mx,mn:real; n4:integer ; var y:arr4);
(This procedure gives X-axis of a variable given max&min&# of divisions)
var
    e:real;
    i:integer;
begin
    e:=(mx-mn)/n4;
    y[1]:=mn+e/2;
    For i:= 2 to n4 do
        y[i]:=y[i-1]+e;
    end;
}-----}

begin {main program begins here }
    reset(f1,'fxdata');
    rewrite(f2,'disp13');
    (rewrite(f3,'dispdpl');)
    (rewrite(f4,'dispdil');)
    (rewrite(f5,'dispV13');)
    (rewrite(f6,'dispf11');)
    (rewrite(f7,'dispLi');)

    writeln(' enter the no. of values required ');
    readln(n1);
    writeln(' Enter the no. of divisions for distribution ( 61 :)');
    readln(n2);
    writeln(' Enter the swelling in % ');
    readln(fs);
    fs:=fs/100;

    m1:=0.01427; s1:=0.00001333;
    m2:=0.0144 ; s2:=0.00002974;
    m3:=0.4858 ; s3:=0.00002 ;
    m4:=55000 ;

```

```

m5:=0.00005; s5:=0.000001 ;
m6:=13.412*270/(238*19); s6:=0.217*270/(238*3*19);
m7:=10600 ; s7:=0.05*1000;

(Applying thermal expansion below )
m1:=m1*(1+fs/3) ;
m2:=m2*(1+0.000006077*350);
(above mean & std. deviations assigned to all inputs)

i:=0; h:=0; h1:=0; h2:=0;
while not (eof(f1)) do begin
i:=i+1;
readln(f1,g1[i],g2[i]);
end;
(reads normal distribution from input file fxdata )

generate(m1,s1,i,n1,g1,g2,dp);
generate(m2,s2,i,n1,g1,g2,di);
generate(m3,s3,i,n1,g1,g2,L );
generate(m5,s5,i,n1,g1,g2,a );
generate(m6,s6,i,n1,g1,g2,m );
generate(m7,s7,i,n1,g1,g2,d );
(assigns normal distribution to all the input variables )

c:=320.86*86400;
(c=residence time of fuel bundles in reactor(55KW/m & 12500MWday/ton)
Tg:=650 ;(average gap temperature = 650 k ref. AERB data)
R:=8.314 ;(R=universal gas constant =8.314J/mol/K)

fissiongas_release(m5,c,2000,fr1);
fissiongas_release(m5,c,1350,fr2);
fissiongas_release(m5,c,700 ,fr3);
fr:=(fr1+4*fr2+fr3)/6;
V1:=pi*m2*m2*m3/4-(1+fs)*m6/m7;
m8:=0.25*R*((0.23824*0.1)/6.023)*Tg*fr/(V1*101325);
(calculates nominal value for output parameter )

NP:=0 ; Pav:=0 ; Vav:=0 ; fav:=0 ; NV:=0;
for h:=1 to n1 do
begin
fissiongas_release(a[h],c,2000,fr1);
fissiongas_release(a[h],c,1350,fr2);
fissiongas_release(a[h],c,700,fr3);
f[h]:=(fr1+fr2*4+fr3)/6;
(Above is the average fractional fission gas release from the pellet)
fav:=fav+f[h] ;
V[h]:=pi*di[h]*di[h]*L[h]/4-(1+fs)*m[h]/d[h] ;
(volume available for fission gases=inner volume-material volume )
Vav:=Vav+V[h] ;
p[h]:=0.25*R*(0.23824/6.023)*Tg*f[h]/V[h];
(assumes fission gases as ideal gases )
p[h]:=p[h]/101325;(converts the pressure in bars )
if (V[h] <= 0) then begin
p[h]:=350;(introduce a maximum pressure)
NV:=NV +1;
end;
Pav:=Pav+p[h] ;
if ( p[h] > 95 ) then NP:=NP+1 ;
end;
(calculates different sets of output for each set of input)

Vav:=Vav/n1 ;
fav:=fav/n1 ;
Pav:=Pav/n1 ;
NP:=NP*100/n1 ;

```

```

distribution(n2,n1,dp,e1,mxdp,mndp);
distribution(n2,n1,di,e2,mxdi,mndi);
distribution(n2,n1,L,e3,mxL,mnL);
distribution(n2,n1,a,e5,mxa,mna);
distribution(n2,n1,f,e8,mxf,mnf);
(distribution(n2,n1,V,e7,mxV,mnV);)
distribution(n2,n1,p,e6,maxp,minp);
(calculates distribution for all input & output parameters)

graphdata(mxdp,mndp,n2,dpg);
graphdata(mxdi,mndi,n2,dig);
graphdata(mxL,mnL,n2,Lg);
graphdata(mxa,mna,n2,ag);
graphdata(mxf,mnf,n2,fg);
(graphdata(mxV,mnV,n2,Vg);)
graphdata(maxp,minp,n2,pg);
(Above X-axis of the input & output variables is calculated)

mxf:=mxf*100 ;(converting to percentage)
mnf:=mnf*100 ;(converting to percentage)
fr:=fr*100;(converting to percentage )
fav:=fav*100;(converting to percentage)

( for h1:=1 to n1 do begin
(write (f2,p[h1]:4:4);
(writeln(f2);
(end;
(writeln(f2);
(writeln(f2,'Below is the distribution of the pressure ');)
(writeln(f2);
(writeln(f2,'fuel rod internal pressure ');)
(writeln(f2);
for h2:=1 to n2 do begin
writeln(f2,pg[h2]:7:2,' ',e6[h2]:5);
(writeln(f3,dpg[h2]:9:7,' ',e1[h2]:5);)
(writeln(f4,dig[h2]:9:7,' ',e2[h2]:5);)
(writeln(f5,Vg[h2]:9,' ',e7[h2]:8);)
fg[h2]:=fg[h2]*100 ;(converting to percentage)
(writeln(f6,fg[h2]:6:3,' ',e8[h2]:5);)
(writeln(f7,Lg[h2]:8:6,' ',e3[h2]:5);)
end;
(writes output distribution in the corresponding files )

writeln(f2,'maximum pressure ',maxp:7:2,' bars');
writeln(f2,'minimum pressure ',minp:7:2,' bars');
writeln(f2,'nominal pressure ',m8:7:2,' bars');
writeln(f2,'average pressure ',Pav:7:2,' bars');
writeln(f2,'% of cases where pressure > coolant pressure =',NP:5:2,' X');
writeln(f2,'No. of time volume vanishes= ',NV);
(writeln(f5,'maximum volume available for gases ',mxV:9,'cu.m');)
(writeln(f5,'minimum volume available for gases ',mnV:9,'cu.m');)
(writeln(f5,'nominal volume available for gases ',V1:9,'cu.m');)
(writeln(f5,'average volume available for gases ',Vav:9,'cu.m');)

(writeln(f6,'maximum fractional fission gas release ',mxf:6:3,' X');)
(writeln(f6,'minimum fractional fission gas release ',mnf:6:3,' X');)
(writeln(f6,'nominal fractional fission gas release ',fr:6:3,' X');)
(writeln(f6,'average fractional fission gas release ',fav:6:3,' X');)

end.

```

Table-1

FUEL BUNDLE DATA

1.0 GENERAL

1.1	Length of the bundle	:	495.3 ± 0.75 mm
1.2	Diameter of the bundle	:	81.74 mm (maximum)
1.3	Weight of the bundle	:	16.59 Kg
1.4	No. of bundles per channel	:	12 (10.1 in active core)
1.5	No. of bundles in reactor	:	12 x 306 = 3672

2.0 DIMENSIONS

2.1 UO2 Pellets

a)	Outer diameter	:	14.27 - ±.04 mm
b)	Length	:	17.2 ± 0.13 mm
c)	Dish depth	:	0.41 - 0.61 mm
d)	Land width	:	0.13 - 0.64 mm
e)	Stack length	:	481.18 + 0.00mm - 2.30mm

f) Number of pellets per  
element

:  $\frac{28 + 3}{- 2} ?$

## 2.2 Sheath

a) Outside diameter :  $15.22 \pm 0.04$  mm

b) Wall thickness :  $0.41 + 0.04$   
- 0.03

c) Length :  $485.8 \pm 0.05$  mm ?

## 2.3 End Plug

Thickness : 4.57 mm (minimum)

## 2.4 Split Spacer Pads

a) Length :  $8.6 \pm 0.4$  mm

b) Width :  $2.5 \pm 0.25$  mm

c) Thickness :  $0.61 \pm 0.03$  mm

d) Number : 72

## 2.5 Bearing Pads

a) Length :  $33.0 \pm 0.4$  mm

b) Width :  $2.5 \pm 0.25$  mm

c) Thickness - middle :  $1.02 \pm 0.05$  mm

- Outer : 1.30  $\pm$  0.05mm.
- d) Number - middle : 12
- Outer : 24

## 2.6 End Plates

- a) Diameter : 68.02  $\pm$  0.05 mm
- b) Thickness : 1.57  $\pm$  0.05

## 2.7 Element Assembly

- a) Pellet to sheath diameter clearance : 0.038 - 0.13 mm.
- b) Concentrated axial clearance : 1.20 - 3.80 mm.
- c) Element to element spacing :
  - a) nominal : 1.22 mm.
  - b) minimum : 0.89 mm.

## 3.0 MATERIAL VOLUMES PER BUNDLE (in cubic mm) :

- 
- 3.1 UO<sub>2</sub> Pellets : 1442.97x10<sup>3</sup> (Min.)
  - 3.2 Sheathing : 178.67x10<sup>3</sup>
  - 3.3 End plugs : 24.55x10<sup>3</sup>
  - +

3.4	End plates	:	5.04x10 <sup>3</sup>
3.5	Split spacer pads	:	0.855x10 <sup>3</sup>
3.6	Bearing pads	:	3.249x10 <sup>3</sup>
3.7	Total zircaloy	:	212.364x10 <sup>3</sup>
3.8	Coolant	:	957.00x10 <sup>3</sup>
3.9	Volume Displaced by bundle	:	1706.13x10 <sup>3</sup>
4.0	MATERIAL WEIGHT (PER BUNDLE) :		
4.1	Uranium oxide	:	14.97 Kg (min.)
4.2	Uranium	:	13.412 Kg. (min)
4.3	Sheathing	:	1.174 Kg.
4.4	End plugs	:	0.161 Kg.
4.5	End plates	:	0.033 Kg.
4.6	Split spacer pads	:	0.006 Kg.
4.7	Bearing pads	:	0.021 Kg.
4.8	Total zircaloy	:	1.395 Kg.
4.9	Ratio of zircaloy to UO <sub>2</sub>	:	0.09
5.0	CROSS SECTIONAL AREAS (Bundle mid plane) (in sq. mm)		
5.1	Uranium Dioxide	:	3030.40

5.2 Zircaloy : 549.94

5.3 Coolant : 1779.54

5.4 Ratio fuel/Zircaloy : 5.51

5.5 Ratio fuel/coolant : 1.70

6.0 PERIMETER (Bundle mid plane)

6.1 Sheath and pads : 1020.81 mm

6.2 Coolant tube : 259.94 mm

6.3 Total wetted perimeter : 1280.75 mm

6.4 Equivalent hydraulic Diameter: 5.56 mm

7.0 REACTOR DATA

7.1 Cooling Water-Condition

a) Inlet temperature : normal : 249 deg.C

shutdown : 262 deg.C

b) Outlet temperature : normal : 293 deg.C

maximum : 298 deg.C

- c) Inlet header pressure : normal : 99.3 kg/cm<sup>2</sup>  
 (Gauge pressure) (9.74 MN/m<sup>2</sup>)  
 maximum : 105.1 kg/cm<sup>2</sup>  
 (10.31 MN/m<sup>2</sup>)
- d) Outlet header pressure : normal : 87.0 kg/cm<sup>2</sup>  
 (Gauge pressure) (8.53 MN/m<sup>2</sup>)  
 maximum : 93.8 kg/cm<sup>2</sup>  
 (9.20 MN/m<sup>2</sup>)
- e) Coolant mass flow in : nominal : 13.9 kg/sec.  
 maximum rated channel
- f) Maximum coolant velocity : 9.33 m/sec.
- g) Coolant tube inside diameter: 82.55 + 0.38 mm  
 - 0.00
- h) Temperature rise per : nominal : 44.4 deg.C  
 channel  
 maximum : 50 deg.C

2 Fuel movement :

- a) Bundle residence time : maximum : 4 years  
 average : 2 years
- b) Velocity of movement : 51 mm/sec.

3 Loading condition :

- a) Maximum hydraulic : 365 kg.  
 compression load
- b) Maximum compressive load : 637 kg.

## 8.0 HYDRAULIC PERFORMANCE

- 8.1 Pressure drop for one bundle plus junction : maximum : 42968 N/sq.m
- 8.2 Reynold's Number (Re maximum): 484000 *check*
- 8.3 Prandtl's Number : 1.022
- 8.4 Mass flow rate (maximum) :  $7.24 \times 10^3 \text{ Kg/m}^2 \text{ sec}$

## 9.0 THERMAL PERFORMANCE

- 9.1 Linear Power rating of maximum rated element : time averaged : 50.2 KW/m ( $\int k dT$  of 4.0 KW/m)
- maximum : 57.8 KW/m ( $\int k dT$  of 4.6 KW/m)
- 9.2 Linear power rating of max. rated bundle : time : 871 KW/m  
averaged
- Maximum surface heat flux :  $1051.0 \text{ KW/m}^2 = \frac{5}{\pi d}$
- Maximum volumetric heat generation :  $315 \times 10^3 \text{ KW/m}^3$
- 9.4 Film heat transfer coefficient:  $63.46 \text{ KW/m}^2 \text{ } ^\circ\text{C}$
- 9.5 Gap conductance : 9.0 KW/m<sup>2</sup> °C

9.6 Thermal energy output : Average: 0.529 GJ/KgU  
(6,110 MWD/TeU)

Design

maximum: 1.296 GJ/KgU  
(15,000 MWD/Te U)

9.7 Sheath temperature : maximum: 301 deg.C

9.8 Maximum fuel pellet temp. : Central 1760 deg.C

Surface 480 deg.C

Average 1000 deg.C

9.9 Power output from maximum  
rated channel : 3.2 MW

9.10 Time Averaged Power output  
from maximum rated bundle :

428 KW

9.11 Power ratio : Central element : 1

Mid ring element : 1.108

Outer ring element : 1.435

TABLE 5-1

FUEL BUNDLE DATA SUMMARY

Fuel Material	Sintered UO <sub>2</sub> pellets
Density of UO <sub>2</sub> (2 Deg C)	10.45 to 10.75 gm/cc
Number of pellets per element	~28
Cladding	Zircaloy 2/4
Cladding thickness	0.41 mm
Element outside diameter(nominal)	15.22 mm
Number of elements per bundle	19
Spacing between elements(nominal)	1.22 mm
Spacing between elements(minimum)	0.89 mm
Bundle length	495.3 mm
Bundle diameter(maximum)	81.74 mm
Coolant tube inside diameter (minimum)	82.55 mm
Weight of UO <sub>2</sub> per bundle (min)	14.97 Kgm
Weight of one bundle	16.59 kgm
Weight of fuel in flux	47.0 tonnes UO <sub>2</sub> (41.5 tonnes contained U)

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TABLE 5-2THERMAL AND POWER DATADESIGN MAXIMUM VALUES

Power output per unit length of an element	57.8 KW/m ( $\int Kde=46$ W/cm)
Power output of a bundle	1002 KW/m (483 KW/bundle)
Burnup	201 MW days (15000 MWd/Teu)
Max. cladding surface temperature	308 deg.C
Mean UO <sub>2</sub> temp. in outer elements in maximum design flux	1100 deg.C approx.
Max. temp. within oxide in maximum rated element.	1980 deg.C approx.

For time averaged maximum power

Power output per unit length of max. rated element in max. flux	50.2 KW/m ( $\int Kde = 40$ W/cm)
Power output per unit length of bundle in maximum flux	871 KW/m length (420 KW/bundle)
Power output from maximum rated coolant channel	3.2 MW
Fission energy output per bundle (average)	81.9 MW days (at 6,110 MWd/TeU)
Fission energy output per bundle (maximum)	130.7 MW days (at 9,750 MWd/TeU)
Surface heat flux (maximum)	150.0 W/sq.cm (330,000 Btu/hr sq.ft)

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TABLE 5-3

FUEL ENVIRONMENTAL CONDITIONSCooling Water Nominal Conditions:

Flow in maximum rated coolant tube	13.9 kg/sec
Net flow area in coolant tube	19.2 sq.cm
Maximum coolant velocity through fuel	9.33 m/sec
Temperature rise in all coolant tubes (maximum)	44.4 Deg.C
Inlet temperature	249 Deg.C
Temperature throughout at shutdown	262 Deg.C
Pressure at inlet to coolant tube	Nominal 92.5 Kg/sq.cm Maximum 102.0 kg/sq.cm
Pressure drop across the 12 bundles in the maximum rated channels	Nominal 6.5 kg/sq.cm

Fuel Movements

Average residence time	2 years
Maximum (nominal) residence time	4 years
New bundles per day when operating	9.6

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